MPR ASSOCIATES, INC.

NINE MILE POINT UNIT 1

LEAK-BEFORE-BREAK ANALYSIS OF HIGH ENERGY PIPING SYSTEMS

MPR-820

Prepared for:

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I. INTRODUCTION

Analyses have been performed to evaluate the likelihood of unstable ruptures in high energy piping at the Nine Mile Point Unit 1. Specifically, the analyses demonstrate that, for representative high energy piping systems in the reactor and turbine buildings, leaks will develop before flaws can grow to unstable sizes, and that the resulting leakage can be detected and appropriate action taken before the risk of unstable piping failure develops.

The objectives of this study are to:

- Identify for analysis representative high energy piping systems, fabricated from both carbon steel and stainless steel and carrying both subcooled water and steam.

- Evaluate existing leakage detection capability at Nine Mile Point Unit 1, and establish a leak rate for both the reactor building and turbine building that is clearly detectable.

- Develop, and benchmark against existing test data, a thermo-hydraulic model for prediction of leak rates through tight cracks in pipes and establish flaw sizes for each piping system that will, under pressure loading only, result in the established detectable leak rate.

- Perform finite element stress analyses of each piping system and evaluate the stresses from deadwieght, pressure and safe shutdown earthquake loads, i.e., ASME Code Service Level D loads.

- Perform elastic and elastic-plastic fracture mechanics evaluations of each piping system to determine if postulated through-wall axial and circumferential flaws will not show substantial growth as a result of Service Level D loadings. Further, show that large (one-quarter circumference) circumferentially oriented, through-wall flaws are stable under fully plastic loads.

- Establish leakage monitoring requirements to assure the postulated detectable leak rate is detected.

The general methodology for performing the fracture mechanics analyses has been developed by nuclear steam system suppliers, utilities, the NRC and NRC consultants, and has been used in numerous operating plant applications, including Systematic Evaluation Program evaluations. Appropriate analysis guidelines and acceptance criteria were outlined in the enclosures to the NRC letter, Reference 1.

II. SUMMARY

Stress analyses and fracture mechanics evaluations have been performed for the main steam, reactor feedwater, emergency condenser steam supply and condensate return and reactor water cleanup piping described in Section III of this report. The evaluations were performed using conservative values of crack extension resistance, axial and circumferential postulated through-wall flaw size, and ASME Code Service Level D axial and bending loads. Flaw sizes for linear elastic analyses were selected as those through-wall flaw sizes (axial and circumferential) which will produce leak rates which can be reliably detected by leakage detection systems currently being monitored at Nine Mile Point Unit 1. The flaw size for the extreme plastic load analysis (90° circumferential) was established by the NRC guidelines in Reference 1.

The specific objectives for the fracture mechanics analyses were the following:

- Using linear elastic or elastic-plastic methods, show that insignificant flaw growth occurs under Service Level D loads for flaw sizes dictated by leakage detection capabilities.
- For extreme plastic loads, show that 90° through-wall circumferential flaws display no unstable tearing behavior.

The results of the evaluations show that no growth of the postulated flaw occurs under Service Level D loads in the main steam, reactor water cleanup, reactor feedwater and emergency condenser condensate return piping. The flaw growth in the emergency condenser steam supply piping is

insignificant (0.17 inches with a 7.94 inch initial flaw). Further, no unstable tearing occurs in any of the piping systems with a postulated 90° flaw under Service Level D loadings. Loads resulting in unstable tearing range from 1.17 to 2.09 times the conservative Service Level D loads assumed in the analysis. Loads resulting in plastic collapse for the postulated 90° flaw are generally greater than the loads required for unstable tearing. Because of the conservatism with which flaw size and Level D service loads were established, and the acceptable results obtained using these conservative criteria, it is concluded that the probability for a catastrophic pipe failure is insignificantly small. Therefore, a full double-ended pipe break need not be postulated as a design basis for defining loads at Nine Mile Point Unit 1.

III. METHODOLOGY AND RESULTS OF ANALYSES

A. Piping Systems

Leak-before-break analyses were performed for piping systems in the Nine Mile Point Unit 1 Reactor and Turbine Buildings: Main Steam, Reactor Feedwater, Emergency Condenser and Reactor Water Cleanup. These systems are considered to have the highest potential for unacceptable break consequences and represent both large and small diameter piping, carbon and stainless steel material, and carry both steam and subcooled water.

B. NRC Guidelines for Analyses

The NRC, as part of its resolution of high energy line break safety issues, provided general guidelines, Reference 1, for using fracture mechanics methods to evaluate leak-before-break conditions in piping in nuclear facilities in the Systematic Evaluation Program. These guidelines can be summarized as follows:

- Demonstrate the capability to detect a 2t (two times the wall thickness) flaw under normal operating conditions.

- Show, using elastic or elastic-plastic fracture mechanics methods, that longitudinal or circumferential through-wall flaws 4t in length will not extend when subjected to ASME Code Service Level D loads. If extension is predicted, show it is insignificant.

- Show that there is margin against unstable growth for a 90° circumferentially oriented flaw subjected to extreme loads.

- Show that there is a positive tendency for part-through wall cracks to grow radially through-wall rather than to grow around the pipe circumference.

The methods used to perform leak-before-break evaluations of the four piping systems at Nine Mile Point Unit 1 differ in one respect from these guidelines. The detection of leakage from a 2t flaw may be impossible for small pipe, as in the reactor water cleanup system, where the 2t flow rate could be hundredths of a gallon per minute. Requirements for finding such small leaks by sensitive local leak detection methods may be extensive and complicated, especially when considering that larger flaws could be tolerated safely. Therefore, it was concluded that the postulated flaw size in each piping system should be based on the leakage rate which can be readily detected by existing leak detection and other methods. As will be shown in Section III.C., a leak rate of one gallon per minute is easily detected by existing floor drain sump pump monitoring methods. Therefore, the flaw size for crack extension studies was established as the one gallon per minute flaw size plus 2t, in accordance with the intent of the NRC guidelines to provide some flaw size margin in the analysis. While this approach differs from the NRC guidelines, it results in the evaluation of substantially larger postulated flaws which can be readily detected. Subsequent fracture mechanics analyses, which are intended to demonstrate stable behavior of these postulated flaws, are therefore more conservative than required by NRC guidelines.

C. Leak Detection

An evaluation of the reactor building and turbine building floor drain sump logs taken in January and February 1984, indicates that a leak rate of one gallon per minute or greater would be easily detected from sump pump run time data, which is taken daily. Table 1 shows sump pump flow rate data for this period. These flow rates are all less than the 1440 gallons per day that would result from a 1 gallon per minute leak.

TABLE 1

SUMP PUMP FLOW RATES
JANUARY-FEBRUARY 1984

Sump Identification	Location	Average Flow Rate (gal/day)	Maximum Flow Rate (gal/day)
R-11	Reactor Building	31.0	42.9
R-12	Reactor Building	5.4	9.4
R-13	Reactor Building	1.1	8.6
R-14	Reactor Building	0.0	0.0
R-15	Reactor Building	412.7	744.4
R-16	Reactor Building	1.5	10.3
T-11	Turbine Building	186.9	336.0
T-13	Turbine Building	20.1	103.9
T-14	Turbine Building	215.8	550.7
T-16	Turbine Building	273.2	783.7
T-17	Turbine Building	14.8	25.1
T-18	Turbine Building	50.0	231.4

Further, an evaluation of existing crack growth rate data indicates that flaw growth does not significantly affect leak-before-break safety margins over a period of several weeks after the one-gallon-per-minute rate is reached. Therefore, no special leak detection equipment is required to detect such a leak before flaw growth becomes significant. On this basis, a one-gallon-per-minute leak rate was established for flaw size calculations in each system.

D. Leak Rate Modeling

The correlation between crack size and leak rate was calculated using CRACKFLO, a specialized computer code developed for this purpose. This computer model assumes that the pressure loss through the crack can be described by a typical fL/D loss mechanism. Choking is evaluated using a homogeneous choking model which depends on local stagnation pressure and stagnation enthalpy at the choke point. The flow area caused by opening of the crack due to internal pressure was determined from formulas given in NRC guidelines in Reference 1. CRACKFLO results compare favorably to measured flows through small slits reported in Reference 2. Conservative estimates of flow through tight cracks were obtained by using a friction factor based on a relative roughness of 0.1. Similar flows are predicted by this model for tight cracks as were reported by the LEAKS 01 model developed for EPRI in Reference 3.

E. Stress Analysis

Finite-element stress analyses were performed for each piping system using the ANSYS computer code. The carbon steel main steam system was modeled from the anchors at the external main steam line isolation valves to the inlets to the turbine stop and control valve manifold. A large portion of the turbine bypass line was also modeled since it affects stresses in the main steam line. The carbon steel high pressure reactor feedwater piping was modeled from the exits of the fifth stage feedwater heaters to the external feedwater isolation valves. The west bank emergency condenser stainless steel steam supply and condensate return lines were modeled in their entirety, between the reactor vessel (steam supply), recirculation line (condensate return) and emergency condensers #111 and #112. Finally, the carbon steel reactor water cleanup piping was modeled from the external reactor water cleanup isolation valve to the first regenerative heat exchanger, including the branch to the inlet of auxiliary cleanup pump No. 1. The five models are shown in Figures 1 through 5. Normal operating conditions and pipe material for each system are shown in Table 2.

The four representative piping systems were analyzed for ASME Service Level D loads: pressure, deadweight, and safe shutdown earthquake. Amplified floor response spectra for seismic analyses were obtained from the bounding analyses of Reference 4. These floor response spectra are based on a Reg. Guide 1.60 ground motion spectra anchored at 0.11g ZPA. This ZPA is in

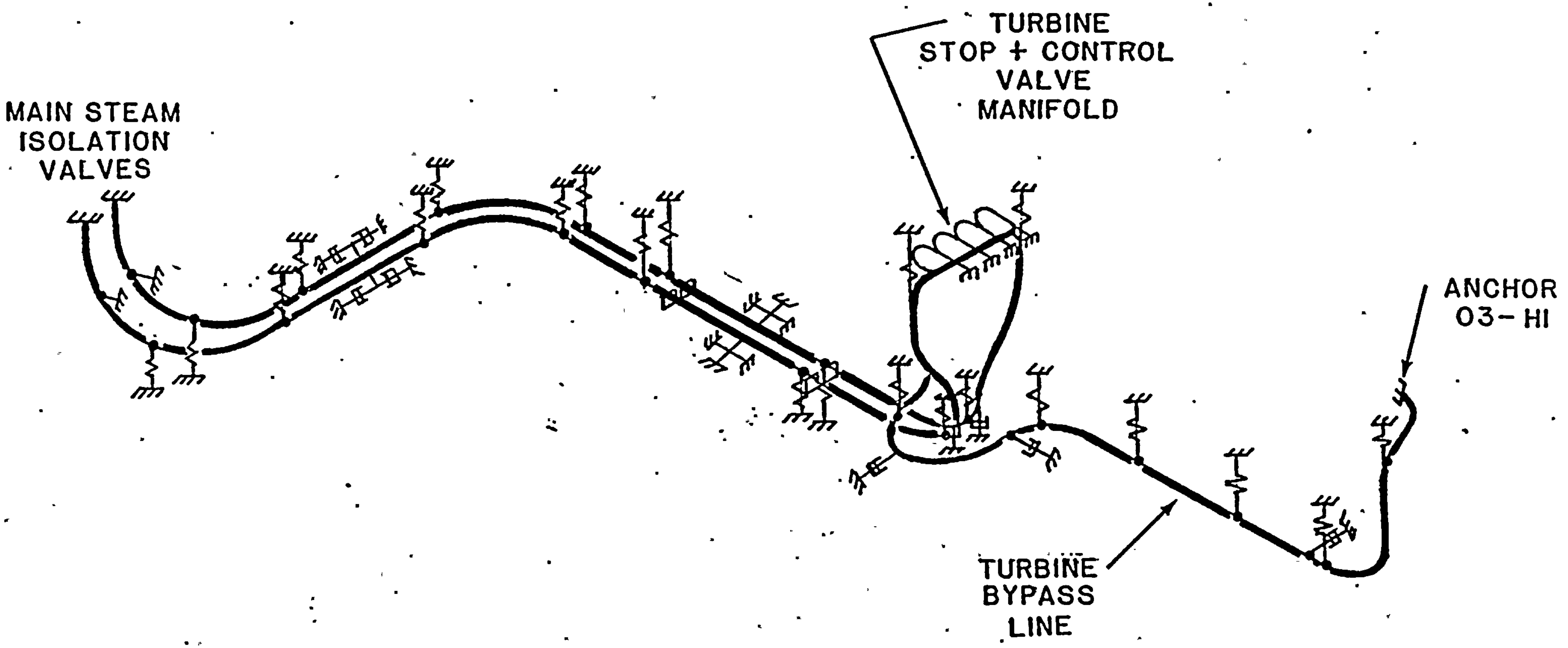

FIGURE I

MAIN STEAM
SYSTEM 02
INCLUDING TURBINE BYPASS TO ANCHOR 03-HI

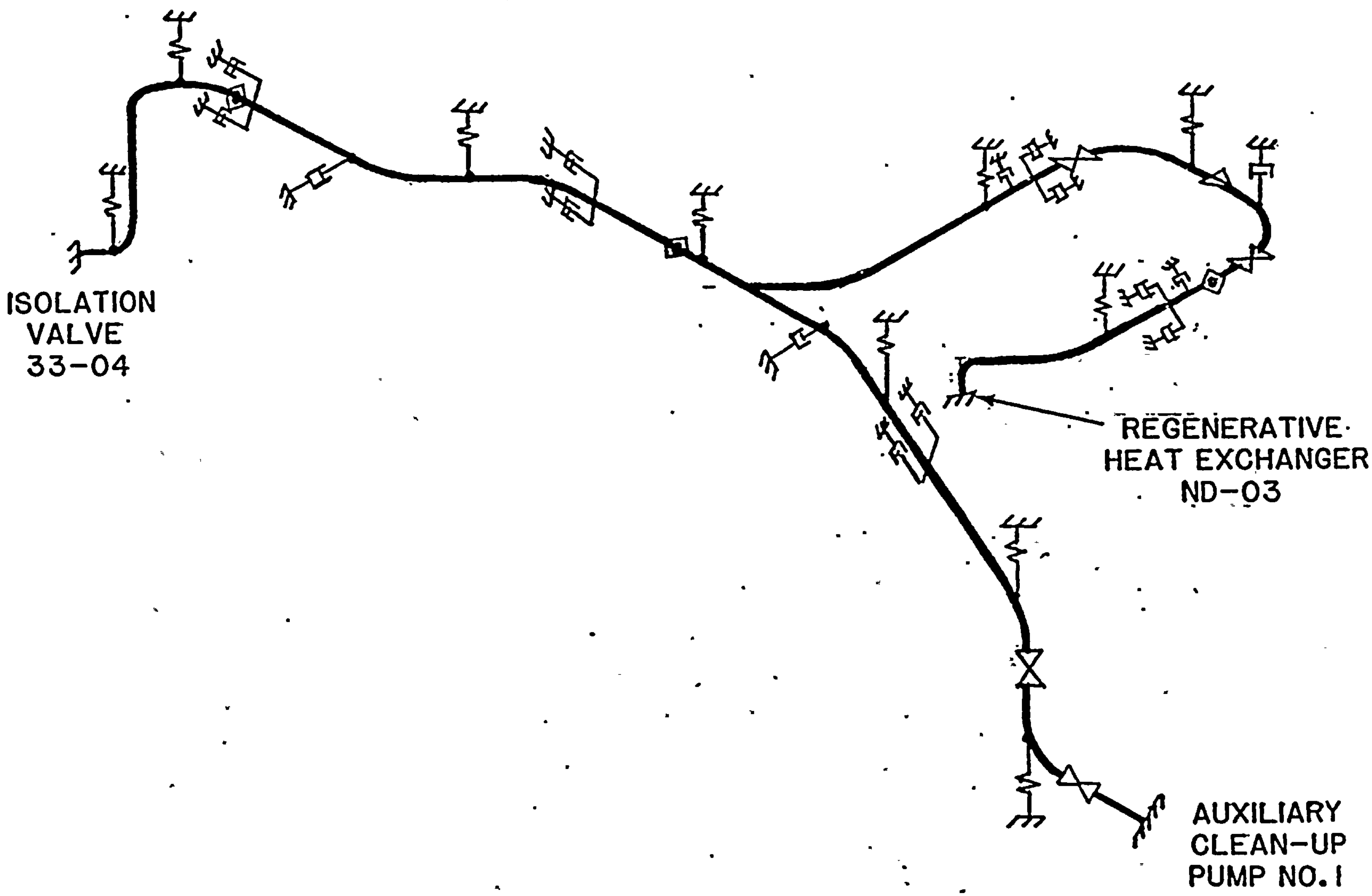

FIGURE 2

REACTOR WATER CLEAN-UP
SYSTEM 33

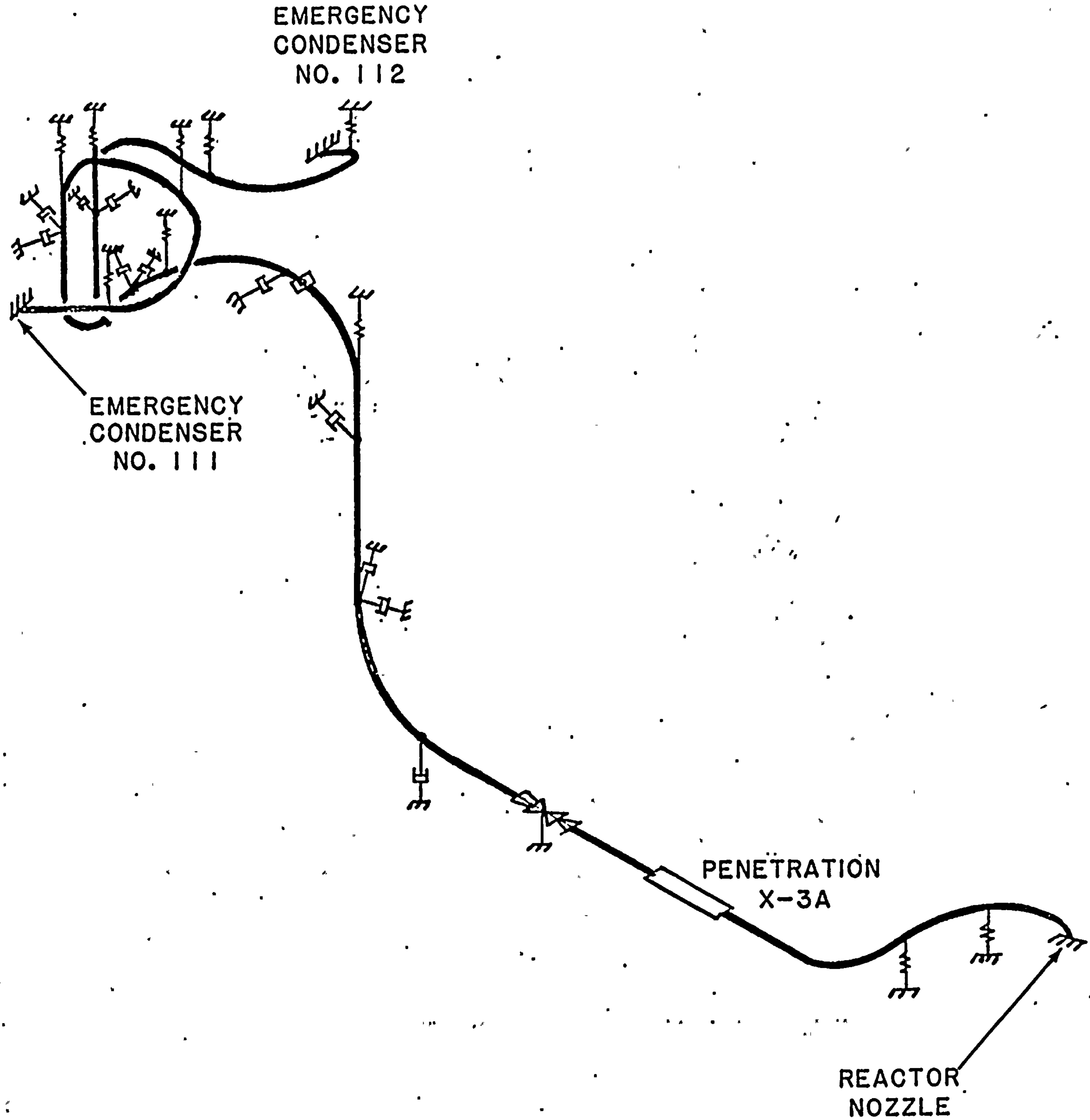

FIGURE 3

EMERGENCY CONDENSER
SYSTEM 39
WEST BANK STEAM SUPPLY

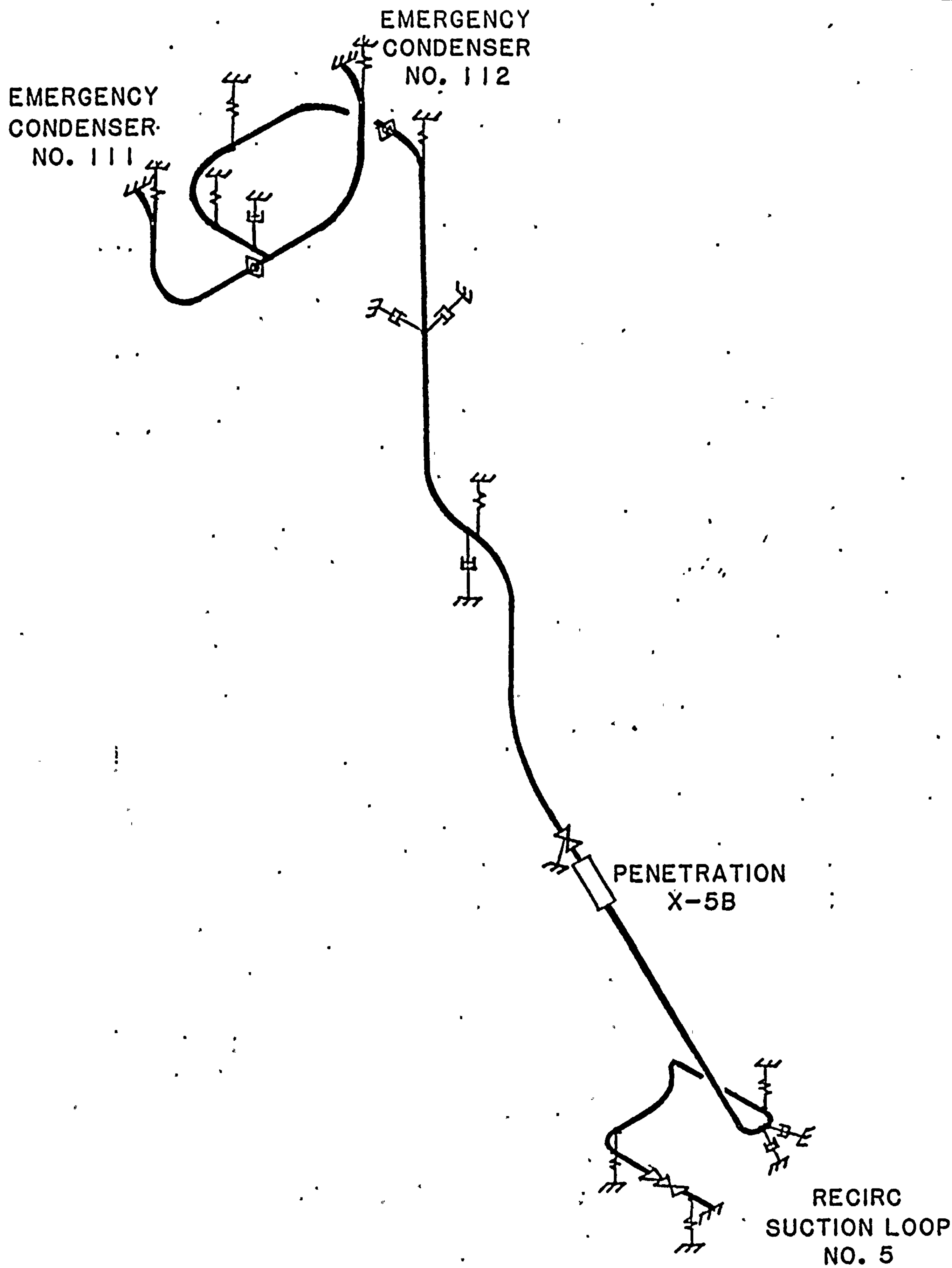

FIGURE 4

EMERGENCY CONDENSER
SYSTEM 39
WEST BANK CONDENSER RETURN

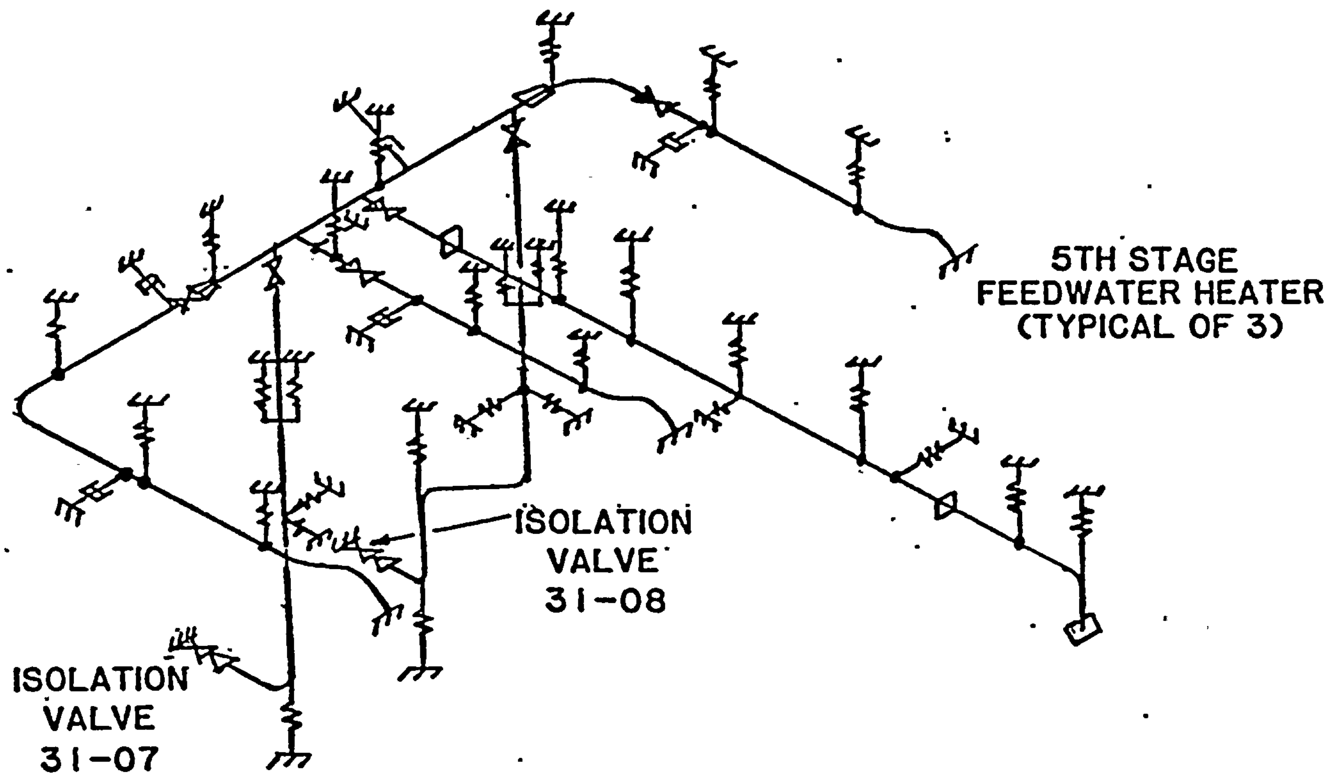

FIGURE 5

HIGH PRESSURE REACTOR FEEDWATER
SYSTEM 30
5TH STAGE FEEDWATER HEATERS TO EXTERNAL ISOLATION VALVES

TABLE 2

OPERATING CONDITIONS

Piping System	Material	Normal Operating Pressure (psi)	Normal Operating Temperature (°F)	Fluid State
Reactor Water Cleanup	Carbon Steel	1030	530	Water
Main Steam	Carbon Steel	1030	550	Steam
Reactor Feedwater	Carbon Steel	1050	360	Water
Emergency Condenser Steam Supply	Stainless Steel	1030	550	Steam
Emergency Condenser Condensate Return	Stainless Steel	1030	530*	Water

* Value for the system in service.

accordance with the NMP-1 design basis given in the FSAR. More realistic response spectra are now being developed for Nine Mile Point Unit 1 as part of the seismic re-evaluation program associated with ISAP (Integrated Safety Assessment Program). The spectra used in the leak-before-break analysis are upper-bound values enveloping the expected final ISAP spectrum. Damping values for the seismic analyses were chosen in accordance with Regulatory Guide 1.61, and are also considered to be very conservative, particularly for the relatively severe Level D loads.

F. Fracture Mechanics Methodology

Elastic and elastic-plastic fracture mechanics analyses were performed for Service Level D loads and extreme loads to evaluate the stability of crack extension in each piping system. The various methodologies are discussed below:

1. Crack Extension - Crack extension under Service Level D loads for the 1 gpm plus 2t flaw was calculated by means of the crack driving potential, or J-integral, as recommended in the Reference 1 guidelines. The calculated value of J at the cracked section is compared to the critical value of J for the material, J_{IC}, to determine if the crack will grow. The determination of J_{IC} for carbon and stainless steels is discussed in Section 3 below.

   When stresses are low, the value of J is related to the more traditional stress intensity factor by the relation

$$J = K_I^2/E.$$

Elastic K_I solutions are available in Reference 1. Suitable plastic zone corrections for ductile materials were applied to calculate an effective crack length, as outlined in Reference 5.

As stresses increase, the net section at the postulated crack can become plastic before J_{IC} is reached. In this case, the more generalized expression for J was used:

$$J = Je + Jp,$$

where Je is the plastic zone corrected value of J discussed above and Jp is the plastic contribution to J. The plastic contribution to J has been studied in detail by General Electric Company in References 6, 7, 8 and 9. The analysis procedure uses methodology for a single edge cracked plate (representing 1/2 the pipe) modified to account for pipe curvature. The plastic contribution to J is expressed in the form

$$Jp = \alpha \sigma_o \varepsilon_o c h_1 (a/b, n) [M/M_o]^{n+1}$$

where

α, n are the strain hardening fitting coefficient and exponent for the material

σ_o is the yield strength, psi

ε_o is σ_o / E

c is the remaining uncracked length on the pipe circumference (for 1/2 the pipe)

a is half the crack length

b is half the pipe circumference (c = b - a)

[illegible]

h_1 is a tabulated function of a/b and n

M is the effective applied moment for Service Level D loads (for 1/2 the pipe)

M_o is the moment at which the remaining uncracked section becomes fully plastic (for 1/2 the pipe)

The plastic contribution to the cracked section hinge angle, φ_c, can be calculated in a similar manner to J:

$$\varphi_c = \alpha \varepsilon_o h_3(a/b,n)\,[M/M_o]^n$$

This function is needed for tearing stability analyses, discussed below.

2. <u>Tearing Stability Analysis</u> - The cracked section resistance to unstable tearing is determined by examining the moment carried by the crack and mathematically perturbing the assumed flaw size. Paris, in Reference 10, states that stability is assured if the moment lost from the cracked section due to an increase in crack length is less than the moment that is picked up by the piping system via the increase in cracked section hinge angle. A stable condition then is represented by:

$$\left|\frac{dM}{d\varphi_c}\right|_{\text{crack}} < \left|\frac{dM}{d\varphi_c}\right|_{\text{piping}}$$

The expression on the right can be evaluated directly from the piping finite element model by inserting into the model a ball joint at the cracked node and applying a moment couple on the joint. In Paris' notation, this is the system

residual stiffness. The system residual stiffness is often equated to the stiffness of a cantilever beam with the same area moment of inertia of the pipe, I, and length L and radius R. The ratio, L/R, of the equivalent cantilevered pipe will be used to report system residual stiffnesses (or compliances) in this report.

The expression on the left can be evaluated in terms of the partial derivatives of J and ϕ_c with respect to crack size, a, and applied moment, M, and a material property called the tearing modulus which is defined as

$$T_{MAT} = E/\sigma_o^2 \frac{dJ}{da}$$

T_{MAT} is determined directly from the slope of J vs Δa test data for the material of interest. In terms of these parameters, the stability criterion becomes:

$$T_{MAT} > \frac{E}{\sigma_o^2} \frac{1}{t} \left[\left(\partial\phi_c / \partial a \right)_M / \left(\partial\phi_c / \partial M \right)_a \right]^2 \times$$

$$\times \left[\left| \frac{dM}{d\phi_c} \right|_{pipe} + 1/\left(\partial\phi_c / \partial M \right)_a \right]^{-1} +$$

$$+ \frac{E}{\sigma_o^2} \left\{ \left. \frac{\partial J}{\partial a} \right)_M - \left[\left(\partial\phi_c / \partial a \right)_M / \left(\partial\phi_c / \partial M \right)_a \right] \left. \frac{\partial J}{\partial M} \right)_a \right\}$$

Since the expressions on the right side of the inequality are all functions of the ratio M/Mo, actual margins to tearing instability can be directly calculated in terms of applied moment.

3. <u>Material Properties</u> - The main steam, reactor feedwater and reactor water cleanup systems of Nine Mile Point Unit 1 are fabricated from A106, Gr.B carbon steel. The emergency condenser steam supply and condensate return lines are A376, type 304 stainless steel.

Strain hardening coefficients for carbon steel were reported for A212 material (similar to A106) in the annealed and also the normalized conditions in Reference 11. Data for A106 material are not available. A conservative upper bound value for n was chosen from among the normalized A212 data. The coefficient, α, was determined from large strain stress strain data for carbon steel in Reference 12.

Large strain stress-strain data are available for Type 304 stainless steel at elevated temperatures in Reference 13. The strain hardening exponent, n, and also α were determined from these data.

The data used to define the J_{IC} and tearing modulus values for A106 Gr.B material were selected as lower bound values from all available data at 550°F, the reactor nominal operating temperature. A strong effect of plate rolling direction on J_{IC} and T_{MAT} was noted, and worst case data were used. Values of J_{IC} and T_{MAT} used in the analysis are from Reference 14.

The data base used to define J_{IC} and T_{MAT} for Type 304 stainless steel was obtained from

stainless steel weld material test data at 550°F in Reference 14. The weld material has lower crack initiation and growth resistance than the base material, and provides a lower bound estimate of material properties.

The material property values used in the analyses are presented in Table 3. In this table, tensile properties for A106 GrB material are based on ASME Code minimum values. The same is true for Type 304 stainless steel except for yield strength, where the value taken is that of the material in Reference 12 whose strain hardening behavior was quantified.

4. Net Section Plastic Collapse - The presence of the postulated 90°, circumferential, through-wall flaw will reduce the ultimate load carrying capacity of the pipe section. To ensure that the cracked section has adequate margin against net section plastic collapse, limit load calculations were performed to define the margin against collapse for extreme bending loads compared to the load at tearing instability.

The limit moment was determined from Reference 9 and is expressed as

$$M_f = 4\sigma_f R^2 t \cdot (\cos\frac{\theta}{2} - \tfrac{1}{2}\sin\theta)$$

where:

R = the mean pipe radius,

t = the pipe wall thickness,

[illegible]

TABLE 3

MATERIAL PROPERTIES

	A106 Gr. B	A376, Type 304
Elastic Modulus, E (ksi)	27.0×10^3	25.6×10^3
Yield Stress, σ_o (ksi)	27.1	23.0
Flow Stress, σ_f (ksi)	43.6	42.0
J_{IC} (in-lb/in^2)	903	992
T_{MAT}	214	182
α	1.94	2.13
n	4.42	3.79

θ = half crack angle

σ_f = flow stress (values given in Table 3)

5. <u>Through-Wall Crack Development</u> - The purpose of this evaluation is to demonstrate that assumed through-wall flaws are appropriate bounding flaws for analysis purposes. Since part through flaws cannot be found by leak detection methods, such flaws must demonstrate a propensity to grow radially and leak before becoming large circumferentially and posing a sudden pipe rupture threat. The preference for radial growth can be demonstrated under normal operating conditions and under conditions of large axial or bending loads.

   For normal operating conditions, a large body of operating history data exist that show that cracks in BWR and PWR primary and secondary systems tend to grow radially and leak before becoming a break threat (References 15 to 17). These data cover various initiation and growth mechanisms and exposure to various stress conditions.

   For large axial and bending loads the crack driving force, J, for a part-through wall crack is always larger in the radial direction than in the circumferential direction (Reference 18). This variation is shown in Figure 6, taken from Reference 18. On the basis that crack driving force will determine the direction of flaw growth, it can be assumed that flaws will grow radially and leak under severe load conditions.

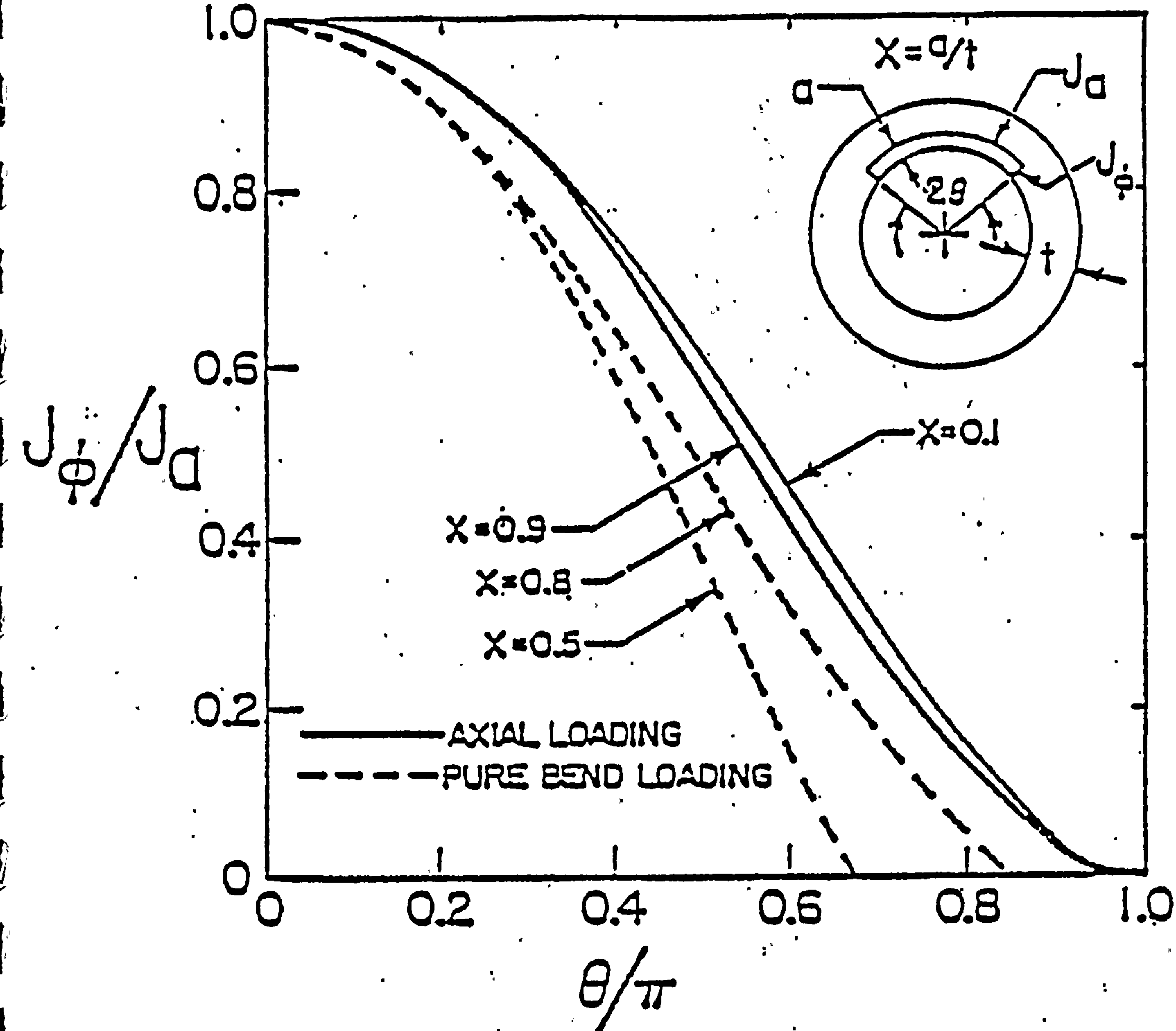

Figure 6

Relative Tendency for Part Through Wall

Cracks to Become Through Wall Cracks

(Reproduced from Reference 18)

G. Results

Numerical calculations were performed to determine stresses in each system resulting from deadweight, pressure and seismic loads. These stresses were combined and the most highly stressed areas in each system were identified.

Through-wall longitudinal and circumferential flaw sizes corresponding to a 1 gpm leak rate were determined for each pipe size in each system. Flaw sizes were increased by two times the wall thickness to provide margin for flaw growth, consistent with NRC guidelines in Reference 1. The resulting flaw sizes corresponding to 1 gpm leak rates in the four systems evaluated are shown in Table 4.

Linear elastic and elastic-plastic analyses were performed in accordance with NRC guidelines in Reference 1 to determine if crack extension was likely to occur under Service Level D conditions. If crack extension was predicted, an estimate of the incremental crack growth was made.

The extreme load case was evaluated for 90° circumferential flaws. The moment carrying capacity of the flawed section was first evaluated assuming infinite system compliance, which is very conservative. This limiting moment was compared with the effective moment (based on total stress) for Service Level D loads to determine margin for unstable crack growth. In some cases, the system residual stiffness was determined at locations of highest stresses, and this effect was

TABLE 4

ONE GPM CRACK LENGTHS

PIPING SYSTEM	OD (in)	1 GPM CIRCUMFERENTIAL CRACK LENGTH A (in)	1 GPM LONGITUDINAL CRACK LENGTH A (in)
Main Steam	16	8.5	6.0
	18	9.5	5.7
	24	8.9	5.0
Reactor Water Cleanup	6.625	4.1	2.5
Reactor Feedwater	14	4.8	3.1
	16	4.9	3.1
	18	5.1	3.2
Emergency Condenser Steam Supply	12.75	6.7	3.8
Emergency Condenser Condensate Return	10.75	3.4	2.1

included in the margin calculation. The plastic limit moment was calculated and evaluated as to whether it would be limiting before the onset of unstable tearing.

1. Fracture Mechanics Evaluation

a. Crack Initiation - Linear elastic and elastic-plastic fracture mechanics analyses were performed to determine if assumed initial 1 gpm + 2t flaws are likely to grow under Service Level D loads. Results for these analyses are shown in Tables 5 and 6. In these tables, it can be seen that $J < J_{IC}$ in every case except for the emergency condenser steam supply piping where a small growth, 0.17 inch, is expected in the circumferentially oriented flaw. This is only 2% of the initial flaw size and is considered insignificant.

b. Extreme Loads - the stability of a 90° circumferential flaw was investigated by varying the applied moment and comparing the moment at instability to the moment resulting from Level D Service loads. This was done for an assumed infinite piping compliance ($L/R = \infty$). In some cases, actual system L/R values were evaluated at locations of highest stress. The values of L/R were calculated conservatively assuming all snubbers and seismic constraints are inoperable, as recommended in Reference 1. The relation between system residual stiffness and L/R is:

$$L/R = \frac{EI}{KR}, \text{ where } K = \left.\frac{dM}{d\phi_c}\right|_{\text{pipe}},$$

E is the modulus of elasticity and I is the area moment of inertia.

Results of this evaluation are shown in Table 7. It is apparent that the margin between the load required for tearing instability and the conservative Level D Service loads is adequate ($>$ 1.0) in all cases. In the case of the minimum margin, 1.17 in the 18-inch reactor feedwater piping, the margin is determined by plastic collapse rather than tearing instability. Tearing instability for this case is not predicted for any load.

TABLE 5

CIRCUMFERENTIAL FLAWS

SYSTEM	OUTSIDE DIAMETER (in)	WALL THICKNESS (in)	MATERIAL	TOTAL STRESS @ FLAW(1) (psi)	A_o(2) (in)	J (in-lbs/in^2)	J/J_{IC}(3)	LEAKAGE FLOW (gpm)	Δa (in)
Reactor Cleanup	6.625	0.432	CS	13,947	5.0	210	0.21	1.6	0
Main Steam	16.0	1.031	CS	16,817	10.56	520	0.58	1.8	0
	18.0	1.156	CS	16,388	11.81	537	0.59	1.9	0
	24.0	1.219	CS	17,708	11.34	470	0.52	2.1	0
Reactor Feedwater	14.0	0.937	CS	20,014	6.67	397	0.44	2.5	0
	16.0	1.031	CS	20,925	6.96	443	0.49	2.7	0
	18.0	1.156	CS	27,026	7.41	688(4)	0.76	3.0	0
Emer. Cond. - Condensate	10.75	0.522	SS	21,336	4.44	417	0.42	2.0	0
Emer. Cond. - Steam	12.75	0.622	SS	23,201	7.94	1,317(4)	1.33	1.7	0.17

NOTES:

(1) Total Stress = Bending Stress + Axial Stress + Pressure Stress under Deadweight + Safe Shutdown Earthquake Loading

(2) One gpm flaw size + 2T

(3) Carbon Steel: J_{IC} = 903 in-lb/in^2

Stainless Steel: J_{IC} = 992 in-lb/in^2

(4) Calculated with elastic-plastic theory.

TABLE 6

LONGITUDINAL FLAWS

SYSTEM	OUTSIDE DIAMETER (in)	WALL THICKNESS (in)	MATERIAL	CIRCUMFERENTIAL PRESSURE STRESS (psi)	A_o (1) (in)	J (in-lbs/in^2)	J/J_{IC} (2)	LEAKAGE FLOW (gpm)
Reactor Cleanup	6.625	0.432	CS	6868	3.36	69	0.07	2.2
Main Steam	16.0	1.031	CS	7097	8.06	130	0.14	3.4
	18.0	1.156	CS	7125	8.01	101	0.11	4.0
	24.0	1.219	CS	9286	7.44	126	0.14	2.7
Reactor Feedwater	14.0	0.937	CS	6794	4.97	36	0.04	4.9
	16.0	1.031	CS	7097	5.16	37	0.04	4.9
	18.0	1.156	CS	7125	5.51	38	0.05	5.2
Emer. Cond. - Condensate	10.75	0.522	SS	9576	3.14	80	0.08	3.4
Emer. Cond. - Steam	12.75	0.622	SS	9527	5.04	Note (3)	Note (3)	2.2

NOTES:

(1) One gpm flaw size + 2T

(2) Carbon Steel: $J_{IC} = 903$ in-lb/in^2

Stainless Steel: $J_{IC} = 992$ in-lb/in^2

(3) Plasticity effects precluded a linear elastic calculation. Value of J expected to be similar to condensate line.

TABLE 7

ELASTIC-PLASTIC RESULTS

SYSTEM	OUTSIDE DIAMETER (in)	WALL THICKNESS (in)	MATERIAL	APPLIED TEARING MODULUS T (1)	LEVEL D STRESS (psi)	MARGIN TO INSTABILITY (2) (3)
Reactor Cleanup	6.625	0.432	CS	31	13,947	2.09
	16.0	1.031	CS	28	16,817	1.74
Main Steam	18.0	1.156	CS	27	16,388	1.79
	24.0	1.219	CS	30	17,708	1.61
	14.0	0.937	CS	30	20,014	1.47
Reactor Feedwater	16.0	1.031	CS	38	20,925	1.40
	18.0	1.156	CS	66	27,026	1.17(4)
Emer. Cond. - Condensate	10.75	0.522	SS	106	21,336	1.25(5)
Emer. Cond. - Steam	12.75	0.622	SS	152	23,201	1.25(6)

NOTES:

(1) Carbon Steel: $T_{mat} = 215$
Stainless Steel: $T_{mat} = 182$

(2) Moment required to unstably tear a 90° flawed pipe divided by the equivalent moment resulting in Level D stress in an unflawed pipe

(3) Unless otherwise indicated, $L/R = \infty$.

(4) $L/R = 110$. For L/R 110, instability is controlled by plastic collapse rather than unstable tearing. For $L/R = \infty$, Margin = 1.09

(5) $L/R = 262$. For $L/R = \infty$, Margin = 1.13

(6) $L/R = 178$. For $L/R = \infty$, Margin = 1.04

IV. REFERENCES

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