REPORT OF INFORMATION PRESENTED AT MAY 13, 1980 NRC MEETING
REGARDING PRELIMINARY RESULTS OF MAIN STEAM LINE BREAK ANALYSES
AND AUXILIARY FEEDWATER SYSTEM AUTOMATION MODIFICATIONS

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I. INTRODUCTION

The information contained in this report is submitted in response to a request to provide the information presented to the NRC at the meeting held on May 13, 1980 to review (1) the various main steam line break (MSLB) accident scenarios and preliminary results performed to date showing a potential for exceeding the containment design temperature and pressure initially reported to you in our April 18, 1980 letter, (2) the details of a system design change to the safety injection actuation logic and the results of the inservice inspection of the main steam line piping being performed during the current refueling outage as a result of the MSLB preliminary results, (3) the evaluation of containment and main steam line integrity which provide assurance that the consequences of such an accident are acceptable and that the probability of a double-ended guillotine rupture of the main steam line is not credible, and (4) the current status of engineering and procurement efforts to implement the modifications to automatically initiate Auxiliary Feedwater System (AFWS) flow to meet the requirements set forth in Item 2.1.7.a of NUREG-0578, "Automatic Initiation of Auxiliary Feedwater System," as documented in our April 29, 1980 letter in response to the request for information contained in your April 3, 1980 letter. The information presented at the meeting was consistent with that provided to the NRC in prior correspondence, except that more detailed information was provided at the meeting. In addition to documenting the information presented at the May 13, 1980 meeting, the report reflects additional analytical work completed subsequent to the meeting.

Section II of the report discusses Items (1), (2) and (3) above, as well as the additional analytical work completed subsequent to the May 13, 1980 NRC meeting. Section III of the report discusses Item (4) above.

Section IV of the report provides a summary of the information contained herein. Based on this information, the following conclusions are made:

1. The preliminary results of the various MSLB accident scenarios obtained to date warrant corrective actions (i.e., modifications to the SIAS logic) to reduce the calculated peak containment pressures. Therefore, the requirement for these modifications constitute a reportable occurrence as defined in Technical Specification 6.9.2.a(9), and as such, is being reported pursuant to the Technical Specifications. In accordance with our April 18, 1980 letter to the NRC Office of Nuclear Reactor Regulation and our May 19, 1980 letter to the NRC Office of Inspection and Enforcement, Region V, a copy of this report is being forwarded to the NRC Region V, together with a License Event Report, as the required narrative material to provide a complete explanation of the circumstances surrounding this matter.

2. The modifications to the SIAS logic have been determined not to involve an unreviewed safety question as defined in 10CFR50.59 or a change to the Technical Specifications; therefore, the implementation of the corrective action is being made without prior NRC approval pursuant to 10CFR50.59.

3. The main steam line piping and containment integrity evaluations, in conjunction with the SIAS logic modifications, performed during the current refueling outage are sufficient to provide assurance that a double-ended guillotine rupture of the main steam line is not credible, that the probable consequences of a MSLB accident are no more severe than the previously calculated peak temperatures and pressures for the containment following a LOCA (considering a best estimate calculation with respect to mass and energy generation and containment heat transfer), and in the event that post MSLB temperatures and pressures exceed those previously calculated following a LOCA, containment integrity is maintained by virtue of the as-built strength of the containment.

II. PRELIMINARY RESULTS OF MSLB ANALYSIS

A. Previous Analyses

The containment structure, including access openings and penetrations, was designed and fabricated to accommodate or dissipate without failure the pressures and temperatures associated with a complete loss of primary coolant. The mass and energy release resulting from a complete loss of primary coolant was considered more limiting than that from a secondary pipe rupture (i.e., a MSLB). Therefore, previous MSLB accidents were only analyzed to determine core response and not containment response. Appendix 1 of this report provides a summary of the previous MSLB accident analyses which indicates that for each of Core Cycles 1 through 8 the results have been shown to be acceptable for core response.

In accordance with the NRC December 21, 1979 letter, the MSLB accident is currently being reevaluated, in conjunction with the NUREG-0578 Lessons Learned Requirement to provide automatic initiation of AFWS flow, to determine the impact of automatic initiation of AFWS flow relative to containment response resulting from this accident. The results of current preliminary analyses to determine the containment response are discussed in Sections II.B and II.C of this report.

B. Current Analyses

1. Introduction

As discussed in our April 18 and 29, 1980 letters, a series of scoping studies have been performed to determine the containment response to a MSLB inside containment. An initial 13 cases, all at full power and all based on preliminary and very conservative mass/energy release data, were examined. These cases reflected the effect of various assumed parameter changes (i.e., earlier main feedwater isolation, AFWS flow and containment spray actuation). Appendix 2 of this report provides a table displaying the results of the initial cases. The results show that a potential for high containment pressure exists at San Onofre Unit 1 given a double-ended guillotine rupture of either the 20" or 24" main steam lines inside containment. A major factor in the calculated break energy release is the large inventory addition resulting from continued main feedwater addition prior to a safety injection signal on low pressurizer pressure. The results of the initial scoping studies prompted a change in system design to achieve a safety injection actuation signal (SIAS) on high containment pressure (2 psig), as well as low pressurizer pressure, to minimize main feedwater addition.

Subsection II.D.1 of this report provides the details of the system design change. An accompanying benefit from this system design change is an earlier containment spray actuation (coincident SIAS and high-high containment pressure of 10 psig).

Following these initial preliminary scoping studies, updated and revised mass/energy release data were obtained that better represent the San Onofre Unit 1 system as modified for early SIAS. The revised data was for both full power and no load operation and was limited to a double-ended rupture of the 24" main steam line common header inside containment as this had previously been shown to be the limiting break. These two power levels were analyzed using both standard assumptions and better estimate assumptions as discussed below. The resulting four cases form the basis for the major results provided in Appendix 3 of this report and discussed in Section II.C of this report and are expected to envelope the containment peak pressure response due to other potential variations in initial station operating conditions and reactor protection system and engineered safeguards response. The effect of varying break areas has been conservatively accounted for by assuming double-ended breaks and dry steam blowdown.

As shown in Figure 1 of this report, the three individual main steam lines (20") from the three steam generators feed into a common header (24") inside containment. Flow measuring venturis (14.32" I.D.) are located in each of the 20" main steam lines between the steam generator nozzle and its connection to the 24" header. The MSLB is assumed to occur anywhere in the 24" common header inside containment. A double-ended break of this line has a total break area of 5.32 ft^2 which results in maximum blowdown from all three steam generators limited by their respective flow venturis with an effective break area of 1.12 ft^2 per steam generator. Steam generator blowdown continues to dry out with continued steaming from auxiliary feedwater addition since San Onofre Unit 1 does not have main steam isolation valves.

2. Major Assumptions

a. Standard Assumptions Cases

The major assumptions of the mass and energy release and containment response analysis are listed and discussed below:

(1) The MSLB break flow is assumed to be pure steam, not two-phase. This is conservative because for MSLB cases with large break areas, steam cannot escape fast enough from the two-phase region of the ruptured steam

generator and the two-phase level rises rapidly to the steam generator nozzle resulting in two-phase blowdown. With two-phase blowdown, part of the liquid in the break flow boils off in the containment and is added to the atmosphere while the rest falls to the sump and contributes nothing to containment pressurization. With a pure steam blowdown, all of the break flow enters the containment atmosphere. The determination of the nature of the break flow requires a detailed entrainment model which is dependent on steam generator characteristics. Such a detailed entrainment model has not been developed for San Onofre Unit 1 steam generators. Therefore, the break flow was conservatively assumed to be dry steam which maximizes the peak containment pressure. The effect and potential benefit of a wet steam blowdown assumption is discussed in Section II.C.

(2) Offsite power is assumed available. Cases assuming a loss of offsite power are less severe than cases where offsite power is available for the reasons discussed in Section II.C.

(3) The most restrictive single active failure is assumed to be the loss of a diesel generator which results in the loss of a safety injection train and a containment cooling train (i.e., containment spray pump.) The effect of other single failures are discussed in Section II.C.

(4) AFWS flow is assumed to be initiated manually at 10 minutes at a flow rate of 250 gpm. The effect of automation of AFWS flow is discussed in Section II.C.

(5) The nominal steam generator mass is calculated for a water level corresponding to the programmed level (percent of narrow range span) plus 5% instrument error (steam generator level) and 8% void fraction uncertainty (full power cases).

(6) The initial power level for full power cases is 103% nominal full load to account for 3% calorimetric error.

(7) A conservative clad surface heat transfer film coefficient is assumed.

(8) Reactor trip is on the steam/feedwater mismatch signal. The SIAS (and main feedwater isolation) is on high containment pressure and containment spray actuation is on coincidence of SIAS and high-high containment pressure.

(9) Main feedwater flow is isolated at 8 seconds on SIAS (high containment pressure). Pumped feedwater flow is based on steam generator depressurization and is ramped to zero during feedwater isolation valve closure. No load feedwater flow is assumed initially to be 5% of nominal full power flow.

b. Better Estimate Cases

The assumptions applicable to the better estimate cases which are different from those above are listed below:

(1) The initial steam generator mass does not include instrument errors or void fraction uncertainty.

(2) The initial power level for full power cases is 100% of nominal full load and does not include 3% calorimetric error.

(3) A more realistic clad surface heat transfer film coefficient is assumed.

(4) Both safety injection trains are assumed operable (failure of a containment spray pump is still assumed).

Conservatisms which remain in the analysis and which were not adjusted for the better estimate cases are listed below:

(1) The most reactive control rod is assumed stuck out of the core.

(2) Primary to secondary heat transfer (UA) across the steam generator tubes is not degraded with decreasing steam generator level.

(3) Steam generator blowdown is assumed to be dry steam (i.e., no entrainment) from an instantaneous double-ended rupture of the largest steam line (24").

(4) Condensate revaporization from containment passive heat sinks is not assumed in the containment response analysis.

3. Mass and Energy Releases to Containment

In the event of a postulated MSLB , high energy fluid is released to the containment, causing an increase in the pressure and temperature of the containment atmosphere. Mass and energy releases to containment were calculated by Westinghouse using the MARVEL code[1] which has been used for previous San Onofre Unit 1 analyses of core response following a MSLB. The mass and energy release to containment were determined from the following sources:

- o Initial steam generator inventory.
- o Steam piping inventory.
- o Feedwater system pumping.
- o Feedwater system flashing.
- o Auxiliary feedwater flow.

a. Initial Steam Generator Inventory

The initial steam generator inventory is a function of the reactor power level. For operation above 20% power, the steam generator operating level is automatically controlled at 30% on the narrow range level instrument. At 20% power and below, the operating level is manually controlled at 50% on the same narrow range level instrument. In addition, steam void fraction below the operating level varies with power strongly affecting the mass inventory. Initial steam generator mass inventories for the four base cases analyzed are tabulated below.

Initial Steam Generator Inventory (lb/S.G.)

	Full Power	No Load
Standard Assumptions	43500	69710
Better Estimate Assumptions	38760	67980

b. Steam Piping Inventory

As shown in Figure 1 of this report, the main steam system piping volume between the steam generators and the main steam stop valves discharges through the break into containment. This volume (1965 ft3) includes branch lines 6" in diameter or above (includes relief and safety valve header, pressure equalizing crosstie, and reheater supply lines with extension to the condenser dump valves.) The mass of steam is calculated based on the density of dry steam at the steam generator pressure for the power level being evaluated and is assumed to precede the reverse flow blowdown from two of the three steam generators. At full power, the steam mass is 3052 lb and at no load the mass is 3979 lb.

c. Feedwater System Pumping

Figure 2 of this report shows the feedwater system pumping configuration. Following a MSLB, feedwater will continue to be pumped into the steam generators until feedwater line isolation occurs as a result of SIAS. For the purpose of this analysis, the SIAS is assumed to occur on high containment pressure. The contribution to the steam generator mass inventory from the main feedwater pumping is determined in the following manner:

(1) The containment pressure and the steam generator depressurization are calculated assuming a conservatively estimated pumped feedwater flow.

(2) From the containment pressure, the time at which the high containment pressure setpoint for a SIAS is reached is determined (less than 1.5 seconds).

(3) The time at which feedwater isolation is complete is based on the time the SIAS setpoint is reached (1.5 seconds), plus instrument response time (1.5 seconds), plus the feedwater isolation valve closure time (5 seconds), or a total time to feedwater isolation of 8.0 seconds.

(4) The feedwater regulating valves are assumed to fully open following a MSLB at full power and remain open until SIAS occurs. On a SIAS, the feedwater regulating valves close (closure time 10 seconds) as a backup to the feedwater isolation valves. At no load, the system is in manual control and the main feedwater addition is limited to a maximum of 5% of full power flow for eight seconds until feedwater train isolation.

(5) The mass of water added to the steam generators prior to feedwater line isolation is based on feedwater pump flow characteristics as a function of steam generator pressure decay until the SIAS is received (3.0 seconds) followed by a linear ramping of the feedwater flow to zero during the time period the feedwater isolation valves are closing (5 seconds) At full power, main feedwater pumping adds 12760 lb while at no load main feedwater pumping adds only 633 lb. These are total inputs, assumed equally divided among all three steam generators.

d. Feedwater System Flashing

Following main feedwater isolation, the depressurization of the steam generators causes flashing of the water in the feedwater piping between the steam generator and the feedwater regulating valves as shown in Figure 2 of this report. The available volume of feedwater piping is 246 ft^3 and is equivalent to a mass of 13030 lb of water at full power based on the feedwater temperature of 417°F. This mass is added to the steam generators where it is boiled off by primary to secondary heat transfer. At no load, the feed train is assumed cool (70°F) and does not flash upon steam generator depressurization.

e. Auxiliary Feedwater Flow

Auxiliary feedwater flow is currently initiated manually following a MSLB. Figure 3 of this report shows the AFWS configuration. For purposes of this analysis, auxiliary feedwater was assumed to be initiated at 10 minutes at a flow of 250 gpm and discharged through the break to containment as dry steam at the same rate it is pumped in following the steam generator blowdown, pumped feedwater addition, and feedwater flashing. The effect of automatic initiation of AFWS flow is discussed in Section II.C.

4. Containment Response Analysis

The containment response analysis was performed by Bechtel using the COPATTA code(2). A detailed description of the containment model (containment initial conditions, heat sink data, heat removal systems) is documented in Enclosure 1 of our January 19, 1977 letter entitled, "Containment Post Accident Pressure Reanalysis, San Onofre Unit 1."

5. References

1. J. M. Geets, "MARVEL, A Digital Computer Code for Transient Analysis of a Multiloop PWR System", WCAP-7909 (Non-Proprietary), June, 1972.

2. Performance and Sizing of Dry Pressure Containments, Bechtel Power Corporation, BN-TOP-3, Revision 4, October, 1977 Draft.

C. Results of Current Analyses

1. Summary

Appendix 3 of this report provides a table which summarizes the results of the four cases run to date that model San Onofre Unit 1 assuming a SIAS on containment high pressure as discussed in Subsection II.D.1 of this report. The four cases consist of two analyses at full power and two at no load. One case each at full power and no load is based on mass/energy (blowdown) data incorporating standard safety analysis assumptions and conservatisms. One case each at full power and no load is also provided based on a nominal, better estimate blowdown data. All four cases assume dry steam blowdown from a double-ended rupture of the 24" main steam header inside containment. Figures 4 and 5 of this report show the containment pressure as a function of time for the full power and no load cases based on the better estimate blowdown (cases 1 and 3) and the standard blowdown (cases 2 and 4), respectively. Figures 6 and 7 of this report show the containment vapor temperature response for the full power and no load cases, respectively, based on the better estimate blowdown data. Containment temperatures are not significantly higher using the standard blowdown data. Included on the temperature plots are curves showing the thermal response of the inside surface of the containment sphere.

Appendices 4 and 5 of this report provide tables which delineate accident chronologies for the full power and no load cases, respectively, based on the better estimate blowdown data out through initiation of AFWS flow at 10 minutes.

As shown in Appendix 3 of this report, the peak containment pressures calculated for double-ended MSLB's at full power and no load, respectively, are 50.0 and 53.0 psig using blowdown data incorporating conservative, standard assumptions. These peak pressures are reduced to 48.4 and 52.2 psig for full power and no load, respectively, when nominal, better estimate blowdown data is employed. It should be recognized that these peak pressures still reflect the significant conservatism of assuming dry steam blowdown from a double-ended guillotine rupture of the 24" main steam line. Blowdown flowrates are initially over three times normal steam flow; substantial steam generator inventory swell and overload of the steam dryers will in fact occur, creating significant water entrainment with resultant lower peak containment pressures expected. The steam flow does not decrease to the nominal full power flow rate (where entrainment would not be expected) until 30 to 40 seconds have elapsed.

The peak vapor temperatures of 403 to 406°F shown in Appendix 3 of this report are of short duration and quickly quenched by the automatically initiated containment spray system. As shown in Figures 6 and 7 of this report, vapor temperatures are below 300°F within 70 to 80 seconds and are at saturation values corresponding to the containment pressure after 90 seconds. The inside surface temperature of the steel containment sphere (modeled as 1" thick carbon steel) is shown on these figures. The sphere surface remains below the saturation temperature throughout the one-hour analysis time reaching maximum values of 239°F with the full power MSLB and 268°F with the no load MSLB.

2. Sensitivity/Parameter Studies

a. The effect of automatic initiation of AFWS flow was assessed as shown in Appendix 2 of this report by rerunning certain cases with the assumption that auxiliary feedwater was initiated at time T = 0 at a runout flow rate of 1000 gpm for 90 seconds (loss of steam pressure drops turbine AFWS pump off line), 500 gpm out to 10 minutes and 250 gpm, thereafter. The early initiation of AFWS flow has the effect of extending steam generator dryout and adding mass and energy to the containment atmosphere. This assumption was found to increase peak containment pressure by approximately 1 psi for the full power cases. For full power cases, the effective delay in AFWS flow initiation (approximately 1 minute) would minimize this effect as steam generator dryout and peak containment pressure would occur prior to significant AFWS flow addition.

b. The effect of wet steam, steam generator blowdown was assessed by rerunning all cases as shown in Appendix 2 of this report using an entrainment model (steam quality vs. time) developed for the Model 51 steam generator (San Onofre Unit 1 has Model 27 steam generators). The benefit of this effect, as discussed earlier, is to decrease the mass and energy available as dry steam which contributes directly to the containment pressure/temperature response. This assumption was found to decrease peak containment pressure by approximately 8 psi for the full power cases. Although a detailed entrainment model has not been developed for San Onofre Unit 1 steam generators, the potential benefit of the more realistic assumption, as indicated from the results of the sensitivity studies, is believed to be significant.

c. The effect of loss of offsite power coincident with the MSLB was qualitatively assessed. The loss of offsite power would result in tripping of the reactor coolant pumps, main feedwater pumps, and delay in AFWS flow addition. Each of these aids in mitigating the effects of a MSLB by either reducing the fluid available to feed the blowdown or reducing the energy transferred from the primary coolant system to the steam generators. The loss of offsite power would result in a slight delay in initiating safety injection flow (10 seconds), containment spray (44 seconds), and main feedwater isolation (10 seconds) due to diesel generator starting delays. However, the backup isolation main feedwater regulating valves and bypass valves will close without additional delay (effective delay in isolating main feedwater is increased from 8 to 13 seconds) and this effect is mitigated by the tripping of the main feedwater pumps. In the cases analyzed with the standard assumptions, the effect of sing 'ailure of a diesel generator (which implicitly assumes ↓oss of offsite power) has been taken into account. Thus, the assumption of loss of offsite power reduces the consequences of a MSLB.

d. The effects of other single active failures were considered. The standard assumptions cases assumed loss of a safety injection train and loss of a containment spray train (equivalent to loss of a diesel generator). The better estimate cases assumed loss of a containment spray train. The failure of a main feedwater isolation valve to close was considered. The failure of a feedwater isolation valve to close will result in additional pumped feedwater being added to the steam generators. However, two backup isolation valves are available (main feedwater regulating valves and main feedwater block valves) to limit the consequences of this assumed failure. The effective feedwater isolation time would only be delayed by 5 seconds (difference between 5 second and 10 second valve closing times) and by assuming this as the single failure, both containment spray trains would be assumed to be operable. Thus, the effect of this or other single failures are not considered to be as limiting as the loss of a diesel generator (standard assumptions cases) or loss of a containment spray train (better estimate cases).

e. The effect of 8% condensate revaporization during the time the containment atmosphere is superheated was assessed by rerunning the better estimate cases. The effect was to reduce containment peak pressure by approximately 1.1 psi for the full power MSLB and produce a negligible reduction in peak pressure for the no load MSLB. The effect of condensate revaporization on peak pressure is strongly dependent on time of occurrence of the peak pressure in relationship to the duration of containment superheat. Peak vapor temperatures were reduced about 15°F with condensate revaporization for both full power and no load MSLB's.

D. Resultant Actions

1. Design Change

As discussed in Subsection II.B of this report, main feedwater flow to the steam generators substantially contributes to the mass and energy released to the containment through the break prior to receiving a SIAS based only on low pressurizer pressure. SIAS initiated on low pressurizer pressure results in securing main feedwater flow to the steam generators within 32-38 seconds by closing the main feedwater pump discharge valves and feedwater flow regulating valves. The resultant main feedwater addition totals 24,000 to 29,000 lbs per steam generator. In order to minimize the mass and energy release to containment contributed by main feedwater flow to the steam generators, the automatic load sequencing system logic is being modified to provide an additional signal to initiate a SIAS upon 2 psig containment pressure as well as low pressurizer pressure. As discussed in Section II.B of this report, the modified SIAS logic will result in securing the main feedwater flow to the steam generators within 8 seconds, substantially reducing the main feedwater addition and the resulting calculated containment peak pressure.

In addition to securing main feedwater flow to the steam generators earlier as discussed above, the modified SIAS logic, in coincidence with a high-high containment pressure of 10 psi, will result in earlier actuation of containment spray to further reduce the calculated containment peak pressure. The preliminary results discussed in Section II.C indicate that containment spray based on the modified SIAS logic will actuate within 30 seconds as compared to within 60 seconds based on the prior SIAS logic.

As discussed above, the automatic load sequencing logic is being modified and will be completed prior to return to power operation following the current refueling outage. As required by the Technical Specifications, the logic modifications have been reviewed by the San Onofre Unit 1 review committee and determined not to involve an unreviewed safety question as defined in 10CFR50.59 or a change in the Technical Specifications. Specifically, the modifications do not (1) increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report, (2) create a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report, or (3) reduce the margin of safety as defined in the basis for any technical specification. In addition, the operability and testing requirements for the automatic load sequencing logic included in the Technical Specifications are not altered. Accordingly, as permitted by 10CFR50.59, the logic modifications are being made without prior NRC approval.

In accordance with 10CFR50.59, the details of the logic modifications, including a written safety evaluation which provides the basis for the San Onofre Unit 1 review committee's determination, have been documented and will be maintained at the station for the duration of the operating license. In addition, a brief description of the logic modifications, including a summary of the safety evaluation, will be provided to the NRC as required by 10CFR50.59 in the annual operating report for San Onofre Unit 1 in accordance with the Technical Specifications. For completeness, this information is provided as Appendix 6 of this report.

2. Main Steam Line Inservice Inspection

An inservice inspection of the main steam line was initiated during the current refueling outage as a measure to provide assurance of steam line integrity in view of the results of the preliminary scoping studies shown in Appendix 2 of this report. The inservice inspection, in conjunction with the leak detection methods discussed in Subsection II.E.2 assure that pipe flaws will be detected and corrective actions initiated prior to the occurrence of a pipe break. A fracture mechanics evaluation was also initiated as discussed in Subsection II.E.1 of this report to show that a pipe break is, in fact, not credible.

The main steam line welding and inspection was performed in 1965/1966. It was fabricated in accordance with Section I of ASME Boiler and Pressure Vessel Code. The original inspection consisted of a 100% volumetric inspection of all pressure welds by means of radiography. Although an inservice inspection of this line is not a code requirement, a 100% radiographic re-examination of the 33 circumferential steam line welds inside containment has been completed during the current refueling outage. This inspection includes the 3 attachment welds to the steam generators.

Minor flaw indications have been located in 6 welds. None of the indications appear to have been service induced. One flaw is being repaired and the remaining five flaws are being addressed by a fracture mechanics evaluation. The disposition of these findings and corrective actions taken will be reported to NRC Region V as required.

E. Evaluations

1. Preliminary Main Steam Line Piping Integrity Evaluation

a. Summary

The objective of this evaluation is to study the integrity of the main steam line for San Onofre Unit 1. The piping integrity evaluation requires three types of calculations; namely, the plastic instability analysis, tearing modulus analysis and the fatigue crack growth analysis. Preliminary estimates of the plastic instability and tearing modulus analyses are discussed below while the fatigue crack growth analysis will be submitted by October 1, 1980.

Based on the Operating Basis Earthquake moments, plastic instability will not occur even in the presence of a 30" thru-wall crack. If Design Basis Earthquake loads are taken into account, it is estimated that flaws as large as 5" to 10" will not become unstable. The effect of a Design Basis Earthquake on the plastic instability analysis will be provided by October 1, 1980.

Preliminary estimates for the tearing modulus analysis indicate that the tearing instability will not result. Additional (J integral) calculations associated with the tearing modulus analysis will be submitted by October 1, 1980 in support of this conclusion.

b. Introduction

Preliminary scoping calculations have been performed to show that a through wall crack larger than an easily detectable size of 1.5 times the thickness of the pipe will not become unstable, and therefore, a double-ended guillotine break will not result. The piping integrity evaluation requires the following three types of calculations.

(1) Plastic instability analysis to show that the piping loads do not exceed the global moment carrying capability.

(2) Tearing modulus analysis to show that the tearing instability will not result.

(3) Fatigue crack growth analysis to show that small initial credible cracks which can be present in any piping system will not become critical after fatigue crack growth due to operating loads.

The plastic instability method predicts the ultimate failure of the piping system even if the crack tears stably to final failure. The definition of instability used here is that a crack will be unstable if it continues to propagate without an increase in load. The general behavior observed in laboratory tests of cracked piping of this type[1,2] is that the crack will propagate stably until the crack is large enough so that internal pressure cannot be maintained. The final determination of the critical flaw size, or that size flaw which would cause piping failure under the loadings considered, is being carried out using a plastic instability analysis method. In order to apply such an analysis procedure, care must be taken to demonstrate that cracks will not become unstable before the general yielding stage is reached. The tearing modulus approach[3] is used for this purpose.

c. Criteria

(1) Global Failure Mechanism - Plastic Instability Analysis

Piping integrity evaluation involves computation of the moment carrying capacity (M_L) of the pipe in the presence of a postulated through wall flaw and showing that the maximum moment, M, on the pipe resulting from severe loading conditions does not exceed the moment carrying capacity (M_L). Therefore, the global criterion for plastic instability is defined as

$$M < M_L$$

This is illustrated in Figure 8 of this report.

The primary concern in the case of an earthquake condition, the worst case loading here, is that of bending loads superimposed on the internal pressure loads already existing in the piping. The case of a circumferentially oriented flaw (which is the case under investigation) in piping geometries can be analyzed using the limit load analysis method (shown in Figure 9 of this report). The method allows consideration of internal pressure and externally applied bending and axial forces. Use of this approach yields the limit moment(7) as:

$$M_L = \frac{4 \quad (\pi - \alpha)^2 R_m^2 \quad t^2 \quad \sigma_f^2 - \pi^2 R_i^4 \quad P^2}{2(\pi - \alpha)^2 R_m^{\ 2} \ t \ \sigma_f} \quad R_o^{\ 2}(2\cos\beta - \sin\alpha)$$

where α = half-angle of crack, in radius (refer to Figure 9 of this report)
P = internal pressure
R_m = mean pipe radius, inches
t = pipe thickness, inches
σ_f = 0.4 (ys u) (flow stress)
σ_{ys} = yield stress
σ_u = ultimate tensile strength
R_i = pipe inner radius, inches
R_o = pipe outer radius, inches
s = singular location of neutral axis (refer to Figure 9 of this report)

This expression was applied to the results of a series of experiments done by Reynolds(2) and the predictions were quite good.

(2) Local Failure Mechanism - Tearing Modulus Analysis

The local mechanism of failure is primarily dominated by the crack tip behavior in terms of crack-tip blunting, initiation, extension and finally crack instability. Depending on the material properties and geometry of the pipe flaw size, shape and loading, the local failure mechanisms may or may not govern the ultimate failure. The ideal fracture criteria should be based on material parameters and analytical capabilities that encompass all these aspects of crack-tip behavior. The tearing modulus approach(3) is attractive in that it accounts directly for the stable tearing which occurs prior to fracture.

The concept of tearing modulus, T, has been developed on the basis of the J-integral resistance curve and the nondimensional quantities T_{mat} and T_{appl}. These quantities are defined as

$$T_{mat} = \frac{E}{\sigma_f^2} \quad \frac{d\,J_{mat}}{da}$$

and

$$T_{appl} = \frac{E}{\sigma_f^2} \quad \frac{dJ}{da}$$

where E = Youngs Modulus
σ_f = Flow Stress
J_{mat} = Value of J integral following the material resistance curve
J = applied value of J

The condition of stability of crack growth is given by the following:

$T_{mat} > T_{appl}$ Stable

$T_{mat} < T_{appl}$ Unstable

when T_{mat} exceeds T_{appl} by a substantial margin, stable crack growth is assured.

Tada et. al.[6] calculated T_{appl} for boiling water reactor (BWR) piping using the following equation and showed that the maximum values assumed by the functions F_1 and F_2 were 1.3 and 0.5, respectively.

$$T_{appl} = F_1 \frac{L}{R} + F_2 \frac{JE}{\sigma_f^2 R}$$

Therefore, a conservative estimate of T_{appl} is given by the following equation which can be evaluated if the J values resulting from different loading conditions are determined by a finite element analysis.

$$T_{appl} = 1.3 \frac{L}{R} + 0.5 \frac{JE}{\sigma_f^2 R} \qquad 1$$

d. Results and Discussion

(1) Global Criterion

Appendix 7 of this report provides a table which shows the calculated limit moment for different assumed through wall flaw lengths. Based on the available piping stress analysis results[4,5], the maximum bending moment is 492,100 ft-lbs which includes the effect of Operating Basis Earthquake loads. Comparison of this bending moment with the limit moments for different assumed flaw lengths presented in Appendix 7 of this report indicates that even a crack length of 30" will not become unstable. If Design Basis Earthquake loads are taken into account, it is further estimated that flaws as large as 5" to 10" will not become unstable, and therefore, a double-ended guillotine break will not result. As previously discussed, the effect of a Design Basis Earthquake will be submitted by October 1, 1980.

(2) Local Criterion

For the main steam line, the J values resulting from different loading conditions are not available; therefore, T_{appl} cannot be explicitly evaluated at this time using Equation 1, above. The local stability criterion is, however, discussed in the following paragraph:

The T_{mat} for the piping material, SA106B, is assumed to be 140 based on unpublished data. The maximum allowable length of pipe L (between supports) to prevent instability is calculated by conservatively assuming two times J_{IC} for the material. The J_{IC} for the material is assumed to be 1100 in-lb/in^2. Therefore, based on an assumed value of J equal to 2200 in-lb/in^2, the pipe length which will cause instability can be calculated at 102.5 ft using Equation 1, above. Use of a J value equal to 4400 in-lb/in^2 (i.e., 4 times J_{IC}) yields a length of 101.7 ft. The maximum unsupported length(8) for the piping under consideration is approximately 27 ft which is significantly less than the critical lengths calculated. Therefore, stability of piping under consideration is ensured.

(3) Scope of Fatigue Crack Growth Analysis

In addition to satisfying the global and local criteria discussed earlier, it is necessary to study the crack growth resulting from operating conditions. As discussed previously, the fatigue crack growth analysis will be submitted by October 1, 1980. The scope of the analysis is discussed in the paragraphs below.

The observed indication (or assumed credible indication) is treated as a sharp crack, and analyzed as to its behavior in future service. Growth due to further cycling is evaluated in fatigue crack growth analyses and then the final flaw size is compared with the critical flaw size for normal, upset and other operating conditions.

The fatigue crack growth analysis will follow the methods specified in ASME Section XI, Appendix A. The operating transients which affect the main steam line will be considered and scheduled over a 40 year period. Initial flaw depths of different credible magnitudes will be analyzed to give detailed information on crack growth behavior. Crack tip stress intensity factors (K_I) will be calculated using an expression for a continuous flaw oriented circumferentially at the inside surface of the pipe. The stresses will be linerized through the pipe wall thickness and will be used to calculate K_I and ΔK_I. The fatigue crack growth for any single transient will be calculated from a crack growth rate law.

e. Conclusions

(1) Global criterion based on limit analysis method is satisfied. Preliminary calculations using Operating Basis Earthquake loads indicate that even a crack length of 30" will not become unstable. The maximum moment including Operating Basis Earthquake is 4.921 X 10^5 ft-lbs as compared to the limit moment of 5.127 X 10^5 ft-lbs for 30" flaw. If Design Basis Earthquake loads are taken into account it is further estimated that flaws as large as 5" to 10" will not become unstable. As discussed previously, the effect of a Design Basis Earthquake will be submitted by October 1, 1980.

(2) Preliminary estimates indicate that the local criterion is satisfied. As discussed previously, additional (J integral) calculations will be submitted by October 1, 1980 to support this conclusion.

f. References

1. Eiber, R. J. Maxey, W. A., Duffy, A. R., and Atterbury, T. J., "Investigation of the Initiation and Extent of Ductile Pipe Rupture", BMI-1856, July, 1969.

2. Reynolds, M. B., "Failure Behavior of Flawed Carbon Steel Pipes and Fittings", GEAP-10236, October 1970.

3. Paris, P. C. et. al., "A Treatment of the Subject of Tearing Instability", Washington University Report NUREG-0311, July, 1977.

4. "Piping Flexibility Analysis," Bechtel Job No. 3246-5, 11/20/64.

5. "Main Steam X and Z Acceleration (Earthquake)", Bechtel Job No. 3246-1. 12-18-64.

6. H. Tads, P. Paris and R. Gamble, "Stability Analysis of Circumferential Cracks in Reactor Piping Systems". NUREG/CR-0838 R1, R5, June, 1979.

7. "Determination of the moment Capacity of Pressurized Piping with Circumferentially oriented Through-wall Flaws", Central File, SM 12.9.1, May 21, 1980.

8. Isometric line drawings Nos. 334530-3, 334532-2 and 334533-2.

2. Leak Detection System

The main steam line piping integrity evaluation discussed above concluded that postulated thru-wall cracks several inches in length would remain stable if subject to the design basis station loading combinations. The thru-wall critical flaw size for the main steam line is estimated to be 5" to 10" in length as discussed in Subsection II.E.1. A thru-wall crack of this length having an equivalent area of one square inch would result in a steam leak rate of 36,000 lb/hr. The capability of the San Onofre Unit 1 leak detection system to detect the resulting leakage to containment associated with such postulated cracks is discussed below.

Detection of leaks from the main steam line to containment is accomplished through the use of any or all of the following methods:

a. Increase in containment sump level and operation of the containment sump pumps.

b. High humidity alarm in containment.

c. High radiation alarm.

With these methods, a leak of one gpm (500 lb/hr) can be detected in a matter of hours. Larger leaks would be detected by indication from process variables (such as steam flow, feed flow, steam generator level, steam generator temperature and pressure) or by high temperature and high pressure alarms in containment.

In the event symptoms of a leak are noted, an investigation would be conducted to determine the source of leakage, including containment entry, if necessary, and an evaluation would be made to determine the proper course of action.

3. Containment Integrity Evaluation

A containment sphere summary stress evaluation was submitted as Enclosure 2 "Summary Sphere Stress Evaluation, San Onofre Unit 1" of our January 19, 1977 letter. Our February 4, 1977 letter revised the material properties provided in Enclosure 2 of our January 19, 1977 letter. Section IV.D of the 1977 evaluation (hereafter referred to as "previous evaluation") concluded that a containment internal pressure of 51 psig evaluated in conjunction with all required stress conditions and load combinations on the containment sphere and on all significant penetrations resulted in stresses within code allowable stresses based on as-built minimum ultimate tensile strength of the containment material.

The previous evaluation was reviewed following the completion of the MSLB scoping studies to determine the maximum pressure capability of the containment sphere based on minimum specification yield strength of the sphere material. As indicated in the previous evaluation (Section IV.B), the stress intensity allowables for the containment sphere are based on minimum specification yield strength or ultimate tensile strength at temperature (5/8 yield strength or 110% ultimate tensile strength divided by four, whichever is less). The yield strength criterion does not govern for the sphere material with respect to code evaluations, but it is an indication of the maximum sphere strength capability and margin available prior to localized yielding which could lead to loss of containment sphere integrity.

The review indicated that in order to achieve the minimum specification yield strength of the sphere material (38,000 psi) an internal sphere pressure of 92 psig would be required. As noted in the previous evaluation (Section IV.B), the as-built material strength exceeds minimum specification. Thus, it is evident that from a structural integrity perspective, the MSLB inside containment can be accommodated with the current design.

Additional assurance of containment integrity was demonstrated by an initial pneumatic integrity test which was conducted for the sphere following construction. The sphere was held at 53.4 psig (115% of design pressure) for one hour. Details of the test procedure and results were submitted to the NRC recently by letter dated April 4, 1980. Thus, further assurance of containment integrity under overpressure conditions has been provided.

In addition to pressure, thermal effects were considered in the previous evaluation as secondary stresses per ASME Section III Subsection NE-3222.2 and as primary loads for the analysis of piping loads on penetrations. Thermal analysis of the containment sphere was originally conducted by Chicago Bridge and Iron Company. The stress analysis was performed for a ΔT of 200°F, based upon a maximum temperature of 271.8°F and an ambient temperature of 72°F. Allowable stresses were in accordance with the applicable sections of the ASME code.

The analysis results given in the previous evaluation (Table 3) for secondary stresses are based upon a maximum sphere temperature of 300°F (Section IV). As noted in the previous evaluation (Section III), the penetration analysis included loads due to thermal growth. In general, stresses in the vicinity of the penetration are less than those in the region of the shell to foundation juncture because the penetration area is reinforced with a doubler plate. Thermal effects on this thickened portion were investigated in the analysis of the penetration. The results indicated that stresses were within allowable limits.

The results of the analysis for the effects of a postulated MSLB indicate an expected maximum temperature of the steel sphere of less than 268°F. Since this is less than the original design basis of 271.8°F and 300°F used for the analysis given in the previous evaluation (Table 3), temperature effects due to a MSLB will be within the allowable limits of 3.0 S_m for secondary stresses.

III. AUXILIARY FEEDWATER SYSTEM AUTOMATION MODIFICATIONS

Our April 29, 1980 letter indicated that the modifications to provide automatic initiation of AFWS flow to meet the criteria of Item 2.1.7.a of NUREG-0578 cannot be completed prior to January 1, 1981. Appendix 8 of this report summarizes the engineering/procurement/analytical completion constraints discussed in our April 29, 1980 letter. As shown in Appendix 8 of this report, the overall engineering design activities, procurement activities, and analytical work are expected to be completed by October 1, 1980, December 12, 1980 and October 1, 1980, respectively.

Based on current engineering activities, preliminary piping and instrumentation diagrams are provided as Figures 10 and 11 of this report showing the AFWS flow configuration and the turbine driven AFWS pump steam side supply configuration, respectively, as proposed to meet the criteria set forth in Item 2.1.7.a of NUREG-0578. As shown in these figures, the AFWS is being modified to establish two AFWS flow trains with independent and separate automatic initiation signals and circuits. Figure 3 of this report shows the current AFWS configuration.

Pending completion of the automatic initiation of AFWS flow, the compensatory measures which will be implemented include:

1. Partial implementation of TMI Lessons Learned Requirements providing remote manual AFWS flow operation from the control room, except following a SIAS.

2. Local manual AFWS flow operation following a SIAS utilizing the dedicated operator stationed at the manual AFWS isolation valves.

A review of previously analyzed transients and accidents indicates that the remote manual/manual operation of the AFWS is acceptable.

IV. SUMMARY AND CONCLUSIONS

The information contained herein documents the current status of analytical, engineering and procurement efforts to implement the post TMI-2 short-term requirement to provide automatic initition of AFWS flow. Based on this information, the following summary and conclusions are made:

A. Previous MSLB accidents were analyzed to determine core response and not containment response. The mass and energy release resulting from a complete loss of primary coolant was considered more limiting than that from a secondary pipe rupture. Therefore, the MSLB accident is currently being reevaluated relative to containment response.

B. Preliminary scoping studies of the MSLB accident were initially conducted to obtain early results using very conservative, simplified assumptions. These results indicated that the peak pressure inside containment may exceed the design basis pressure for the containment. Revised preliminary scoping studies have been performed using conservative, standard (licensing basis) assumptions and better estimate assumptions. In addition, these studies reflect modifications to the SIAS so that an SIAS is generated on high containment pressure to secure main feedwater flow earlier in the accident. The results of the revised studies indicate that the peak containment pressures would be 50 psig (standard assumptions) and 48.4 psig (better estimate assumptions) at full power, and 53.0 psig (standard assumptions) and 52.2 psig (better estimate assumptions) at no load. The results of the revised studies still contain substantial conservatisms with respect to mass and energy generation (i.e., dry steam and constant primary to secondary heat transfer) and containment heat transfer (i.e., no condensate revaporization). A best estimate calculation without these conservatisms would be expected to be no more severe than the previously calculated peak pressures for the containment following a LOCA.

   The calculated peak containment pressures for the full power cases do not exceed the current containment design basis of 51.0 psig (based on as-built material properties) and the better estimate pressure for the full power case does not exceed the peak containment pressure previously reported for a LOCA (i.e., 49.4 psig). For the no load cases, the calculated peak containment pressures are slightly above the containment design basis pressure. However, these pressures are still below the initial test pressure of the containment. Evaluations have been performed to provide assurance that a double-ended guillotine rupture of the main steam line postulated for purposes of this analysis is not credible and that substantial margin in containment strength is available to accommodate these no load pressures. In addition, station operation at low power (i.e., less than 20%) constitutes a small fraction of the total station operating history.

With respect to thermal effects, the preliminary scoping studies indicate that the maximum temperature of the steel containment is expected to be less than 268oF. This is less than the original design basis for the containment.

As indicated in our January 16, 1980 letter, the complete analytical assessment associated with providing automatic initiation of the AFWS flow as requested by the NRC November 15 and December 21, 1979 letters will be submitted by October 1, 1980.

C. The preliminary results of the various MSLB accident scenarios obtained to date warrant corrective actions to reduce the calculated peak containment pressures. Accordingly, the SIAS logic is being modified to occur on high containment pressure. The system design change will result in securing main feedwater flow early in the accident and minimize the amount of additional mass and energy contributed by this flow. This requirement for corrective actions constitutes a reportable occurrence as defined in Technical Specification 6.9.2.a(9), and as such, is reportable to the NRC Region V pursuant to the Technical Specifications.

The system design change has been determined not to involve an unreviewed safety question as defined in 10CFR50.59 or a change to the Technical Specifications. Accordingly, the change is being made without prior NRC approval.

D. A containment integrity evaluation has been performed to determine realistic margins for the pressure retaining capability of the containment. The evaluation indicates that based on minimum specification yield strength (versus one-fourth of 110% of the tensile strength used as a basis for code allowables), an internal pressure of approximately twice (i.e., 92 psig) the design pressure can be accommodated with the current design. Therefore, the containment integrity following a postulated double-ended guillotine rupture of the main steam line piping will be maintained. In addition, the containment was tested for overpressure prior to initial station operation at a test pressure of 53.4 psig for an hour. The test pressure exceeds the calculated peak containment pressures in all cases, thus providing further assurance of containment integrity.

E. A preliminary evaluation of the main steam line piping integrity was performed to determine the credibility of a postulated instantaneous double-ended guillotine rupture of the main steam line. The evaluation indicates that very large postulated thru-wall cracks will not become unstable, thus precluding the possibility of a double-ended guillotine rupture of the main steam line piping. A postulated thru-wall crack of a size which has been evaluated to be stable would be readily detected by inservice inspection or by the station leak detection methods during operation and corrective actions initiated prior to any threat to main steam line piping integrity.

This evaluation is similar to the mechanistic pipe break evaluation currently being reviewed by the Regulatory staff which was conducted for the Westinghouse Asymmetric Loads Owners Group (WCAP-9570). Acceptance of this approach for that application would mean, in effect, that double-ended guillotine ruptures would no longer be postulated for structural analysis.

The final evaluation to determine the integrity of the main steam line piping will be submitted by October 1, 1980.

F. A radiographic inservice inspection of all circumferential steam line weld joints inside containment has been completed. Six flaws were discovered during the inspections and will be satisfactorily dispositioned prior to return to power operation following the current refueling outage.

G. Based on analytical/engineering/procurement constraints, modifications to provide automatic initiation of the AFWS flow cannot be completed prior to January 1, 1981. Until completion, compensatory measures include (1) remote manual AFWS flow capability, except following a SIAS, and (2) local manual AFWS flow capabiliaty following a SIAS. Previous analyses indicate that this manner of operation is acceptable.

Based on the information summarized above, it is concluded that there is reasonable assurance that a double-ended guillotine rupture of the main steam line is not credible, that the probable consequences of a MSLB accident are no more severe than the previously calculated peak temperatures and pressures for the containment following a LOCA (considering a best estimate calculation with respect to mass and energy generation and containment heat transfer), and in the event that post MSLB temperatures and pressures exceed those previously calculated following a LOCA, containment integrity will be maintained by virtue of the as-built strength of the containment.

APPENDIX 1

PREVIOUS ANALYSES

1. Initial MSLB Analysis (Design Basis Accident - Core Cycle 1)

   - o Spectrum of Breaks Analyzed Inside and Outside Containment at No Load and Full Load Conditions

   - o MSLB Analyzed for Core Response and Not Containment Response

   - o MSLB Mass and Energy Release Not Considered Limiting for Containment Design Basis Pressure

   - o Worst Case Break Postulated at Steam Generator B Tee Connection to Main 24-Inch Pipe at No Load Conditions

   - o Core Response Shown to be Acceptable

2. Core Reload MSLB Reanalyses (Core Cycles 2, 3 and 4)

   - o Different Core Parameters

   - o Similar Conclusions as above

3. Core Reload MSLB Reanalyses (Core Cycles 5, 6, 7 and 8*)

   - o Different Core Parameters

   - o Loss of Off-Site Power

   - o Elevated Upper Reactor Vessel Head Temperatures

   - o Reduced Safety Injection System Flow

   - o Four Combinations of Break Sizes Inside and Outside Containment Analyzed For Core Response

   - o Similar Conclusions as above

* Cycle 8 operation will begin following the current refueling outage.

APPENDIX 2

SAN ONOFRE - UNIT 1

MAIN STEAM LINE BREAK ANALYSIS PRELIMINARY RESULTS

Case	Pipe Size (in.)	Steam Condition	M.F.W. Isolation (sec.)	Spray Start (sec.)	Max. Cont. Pressure (psig)	Max. Cont. Pressure (sec.)	Max. Cont. Temp. (^{o}F)	Max. Cont. Temp. (sec.)	Comments
I-A	20	DRY	34.5	56	53.7	93	423	56	Base Case
I-B	24	DRY	31.5	53	62.0	165	432	53	No modifications
I-C	20	WET	35	56.5	47.6	394	340	57	manual Aux. F.W.
I-D	24	WET	38	59.5	60.4	134	351	59	at 10 min.
II-A	20	DRY	8	56	45.4	56	418	56	Early MFW term.
II-B	24	DRY	8	53	56.8	72	432	53	at 2 psig containment
II-C	20	WET	8	56.5	38.3	114	307	56	manual Aux. F.W.
II-D	24	WET	8	59.5	47.4	49	337	49	at 10 min.
III-A	20	DRY	8	56	46.7	60	420	56	Same as Series II
III-B	24	DRY	8	53	57.8	84	432	53	with automated
III-C	20	WET	8	56.5	- - - NOT RUN - - -				Aux. F.W. at max.
III-D	24	WET	8	59.5	49.8	55	344	55	flow thru 10 min.
IV-B	24	DRY	8	33	53.7	72	411	33	Case II-B with min. spray delay

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NOTES: All cases assume double-ended pipe break
Reactor at full power
Loss of one spray pump

WF:wpf

APPENDIX 3

MAIN STEAM LINE BREAK - CONTAINMENT RESPONSE
PRELIMINARY RESULTS

SAN ONOFRE - UNIT 1

Break: 24-in. double-ended guillotine rupture
Break Area: 2 x 2.66 ft^2
Blowdown: dry steam; flow limited by 1.12 ft^2 area flow venturis in each steam generator outlet line.
Offsite Power: available
Safety Injection Signal at 2 psig.
Containment Spray Signal at 10 psig.
Auxiliary Feedwater on at 10 minutes

	Case			
	1	2	3	4
Mass/Energy Release Assumptions	Better Estimate	Standard	Better Estimate	Standard
Reactor Power (%)	100	103	0	0
Peak Containment Conditions				
Pressure (psig)	48.4	50.0	52.2	53.0
Time (Sec)	78	110	355	378
Vapor Temperature (oF)	405	406	403	404
Time (Sec)	32	32	31	31
Energy Integrals @ Peak Containment Pressure (10^6 Btu)				
Break Flow	166.62	190.32	246.17	242.48
Passive Heat Sinks	26.65	35.79	68.24	70.44
Spray Heat Transfer	1.93	2.91	10.01	10.76

APPENDIX 4

CHRONOLOGY OF EVENTS
FOR THE FULL POWER, BETTER ESTIMATE, 24 IN. D. E. MSLB (CASE 1)

TIME (SECONDS)	EVENT
0	Pipe ruptures; steam generator depressurization begins
1	Containment @ 2 psig; SI signal generated
1	Reactor tripped on steam-feedwater flow mismatch
3	Control rods begin entering core; main feedwater pump isolation and control valves begin closing
4.5	Containment @ 10 psig; spray pump start signal generated
8	Main feedwater flow to steam generators terminated
13	Main feedwater control valves closed (back-up to MFW pump isolation valves)
30.1	Containment spray flow reaches nozzles
34.3	Full containment spray flow established (1080 gpm)
78	Containment reaches peak pressure of 48.4 psig
91	Dryout of steam generators
550	Containment sphere inside surface temperature reaches maximum value of 239.2°F
600	Auxiliary feedwater flow established to all steam generators at 250 gpm (total)

APPENDIX 5

CHRONOLOGY OF EVENTS FOR THE ZERO POWER, BETTER ESTIMATE, 24-IN. D.E. MSLB (CASE 3)

TIME (SECONDS)	EVENT
0	Pipe ruptures; steam generator depressurization begins
1	Containment @ 2 psig; S.I. signal generated
3	Main feedwater pump isolation and control valves begin closing
3.5	Containment @ 10 psig; spray pump start signal generated
8	Main feedwater flow to steam generators terminated
13	Main feedwater control valves closed (backup to MFW pump isolation valves).
29.1	Containment spray flow reaches nozzles
33.3	Full containment spray flow established (1080 gpm)
355	Containment reaches peak pressure of 52.2 psig
375	Dryout of steam generators
600	Auxiliary feedwater flow established to all steam generators at 250 gpm (total)
650	Containment sphere inside surface temperature reaches maximum valve of 268.3°F

APPENDIX 6

DESCRIPTION AND SAFETY EVALUATION
MODIFICATIONS TO AUTOMATIC LOAD SEQUENCING LOGIC

DESCRIPTION

This design change incorporates modifications to the automatic load sequencing logic to provide an additional signal to initiate safety injection upon 2 psig containment pressure. Accordingly, a safety injection actuation signal (SIAS) will be generated upon 2 out of 3 pressurizer low pressure (existing) OR 2 out of 3 containment high pressure (new). The containment high pressure signals will be provided by the existing containment isolation actuation system (CIAS).

SAFETY EVALUATION

An evaluation of the logic modifications has been performed to ensure that isolation of the main feedwater flow by initiating SIAS upon containment high pressure has no impact on other safety systems or analyses.

The design provides adequate separation and isolation to ensure that the logic modifications do not impact the integrity of the CIAS which is derived from the same containment pressure sensors (at 2 psig setpoint). Manual initiation and reset of the automatic load sequencing logic and CIAS remain unchanged. Failure of the common power supply to the CIAS associated with each automatic load sequencing logic will not result in automatic load sequencing logic initiation; however, it will initiate containment isolation.

Based on the Technical Specifications, the automatic load sequencing logic is required to be tested every two weeks. During the testing the input SIAS logic OR configuration will be changed to a logic AND. Therefore, with one SIAS sub-channel in test, false actuation of another sub-channel will not cause spurrious automatic load sequencing logic initiation. This capability is provided by logic modifications and a test mode switch installed on each automatic load sequencing logic test panel. While in the test position, the SIAS will be generated based on containment high pressure and low pressurizer pressure. Test switch position indication is provided in the main control room. The testing requirements are consistent with those permitted by IE Bulletin No. 79-06A, Revision 1, with respect to performing surveillance testing of the pressurizer low pressure channels. In order to prevent spurrious automatic load sequencing logic actuation, the pressurizer low level channels (required to be tripped by the Bulletin) may be restored to normal operation for the duration of the test.

The systems and equipment included in the logic modifications were designed, manufactured, and installed as temporary safety grade consistent with other modifications associated with the implementation of Category "A" Lessons Learned Requirements. Upgrading to safety grade, if necessary, will be completed by January 1, 1981.

A review of the accidents and transients previously analyzed has been performed to determine the impact of producing an SIAS on high containment pressure. Based on that review, an earlier SIAS has a beneficial effect on the analyses.

APPENDIX 7

CALCULATED LIMIT MOMENT

Crack Length (inches)	Limit Moment (ft-lbs)
0	1.7216×10^6
3	1.6025×10^6
5	1.5200×10^6
10	1.3073×10^6
20	0.80375×10^6
30	0.5127×10^6

APPENDIX 8

COMPLETION CONSTRAINTS

1. Engineering

- o Sequencial Engineering Activities Necessary to Issue P.O.'s to be Completed by August 1, 1980
- o P.O.'s to be Issued by August 1, 1980
- o Design Details to be Submitted by October 15, 1980

2. Material Procurement

- o 44 Week Normal Delivery/17 Week Expedited Delivery of Copes-Vulcan Flow Control Valves From Receipt of P.O.
- o Based on august 1, 1980 P.O. Issuance, Delivery By December 12, 1980
- o All Other Items Quoted As 12 Weeks or Less Delivery

3. Analytical Assessment

- o MSLB and Feedwater Line Break Information in Accordance with December 21, 1979 NRC letter By October 1, 1980
- o Auxiliary Feedwater Flow Requirements in Accordance with November 15, 1979 NRC Letter By October 1, 1980

4. Other Assumptions

- o Material Delivery Quotations Based on Preliminary, Conceptual Engineering
- o Piping will be Borrowed From San Onofre Units 2 and 3 and will be Field Fabricated
- o Current Design Basis For Auxiliary Feedwater System Flow Verified by Analytical Assessments
- o Abbreviated TMI Design Review Procedures Will Be Utilized
- o Engineering Activities Can Proceed Without Interferences from other TMI Activities or Any New Activities Resulting From NRC Denial to Defer Items to SEP

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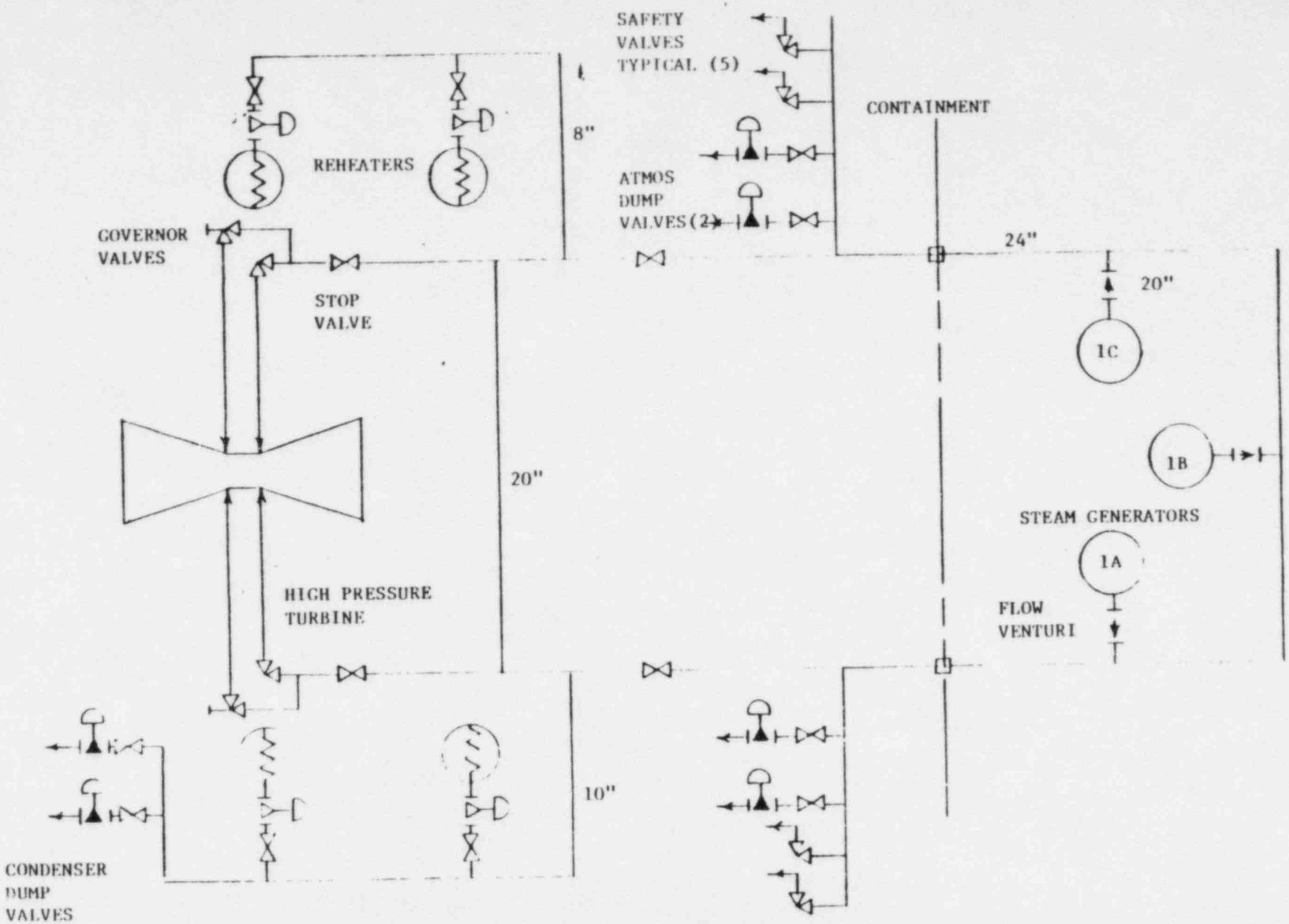


FIGURE 1
MAIN STEAM SYSTEM
SAN ONOFRE UNIT 1

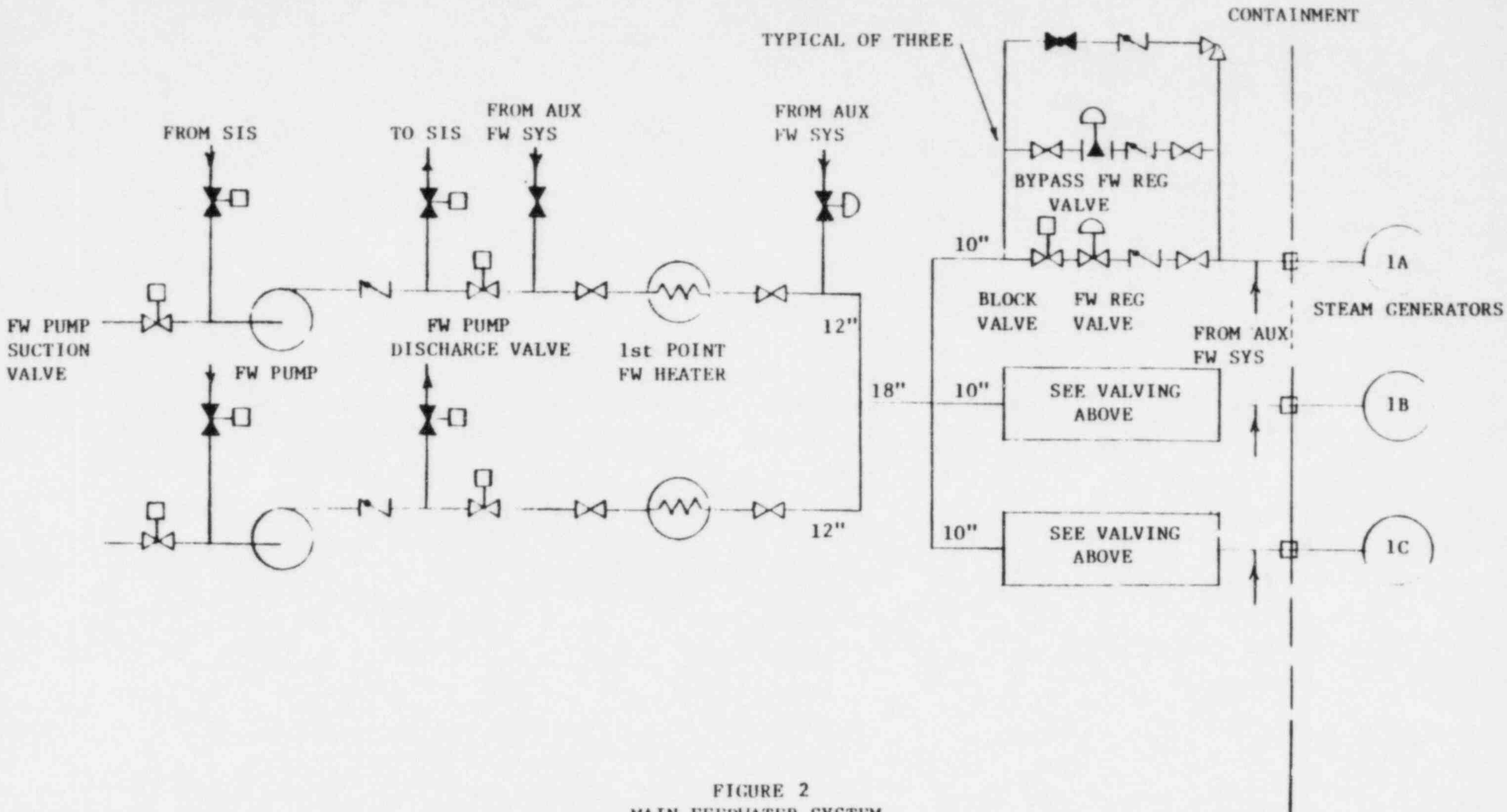


FIGURE 2
MAIN FEEDWATER SYSTEM
SAN ONOFRE UNIT 1

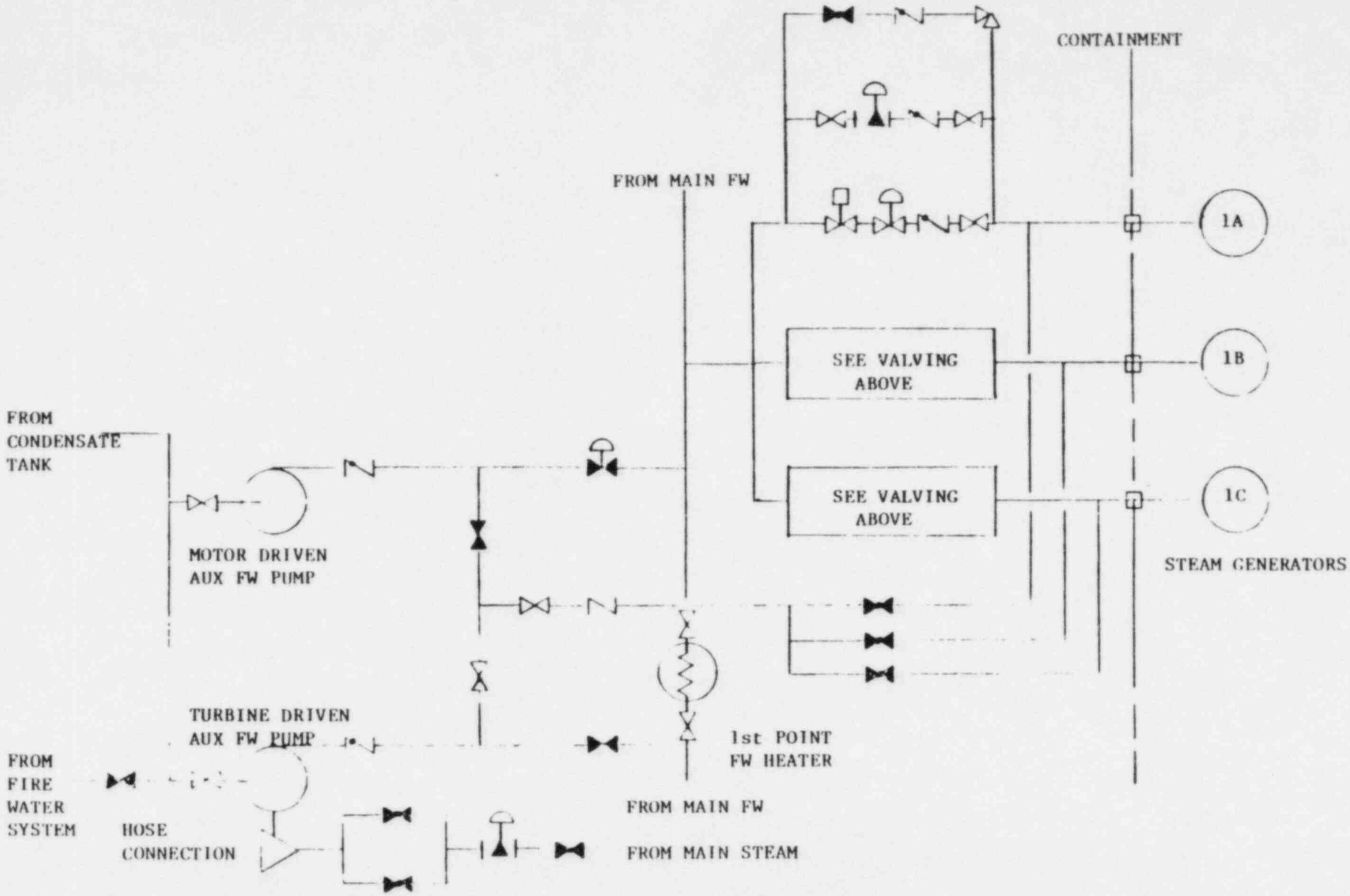
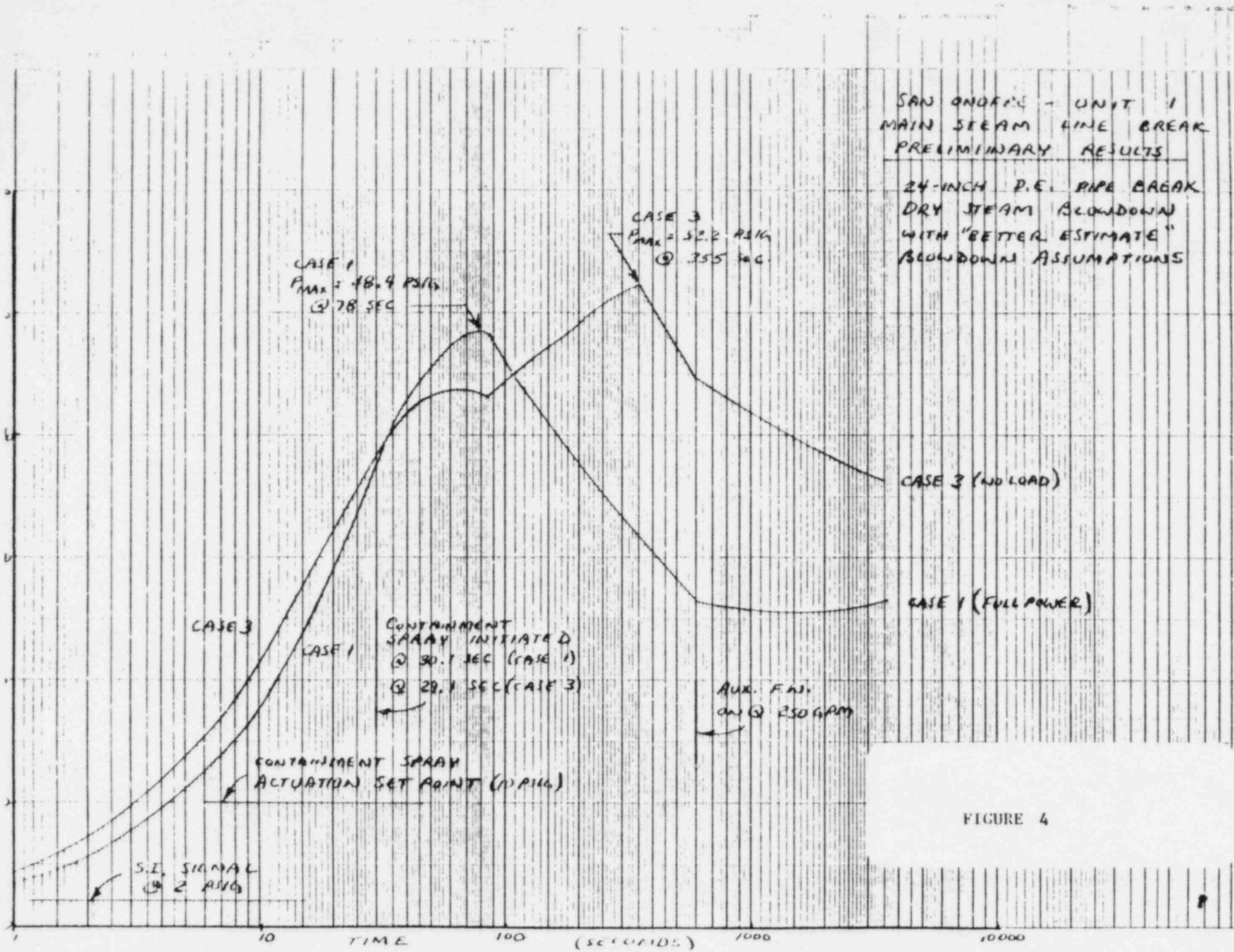


FIGURE 3
AUXILIARY FEEDWATER SYSTEM
SAN ONOFRE UNIT 1



SAN ONOFRE - UNIT 1
MAIN STEAM LINE BREAK
PRELIMINARY RESULTS
24-INCH D.E. PIPE BREAK
DRY STEAM BLOWDOWN
WITH "BETTER ESTIMATE"
BLOWDOWN ASSUMPTIONS
CASE 1
PMAX = 48.4 PSIG
@ 78 SEC
CASE 3
PMAX = 52.2 PSIG
@ 355 SEC
CASE 3 (NO LOAD)
CASE 1 (FULL POWER)
CASE 3
CASE 1
CONTAINMENT SPRAY INITIATED
@ 30.1 SEC (CASE 1)
@ 29.1 SEC (CASE 3)
AUX. F.W. ON @ 250 GPM
CONTAINMENT SPRAY ACTUATION SET POINT (10 PSIG)
S.I. SIGNAL @ 2 PSIG
1
10
100
1000
10000
TIME (SECONDS)

FIGURE 4

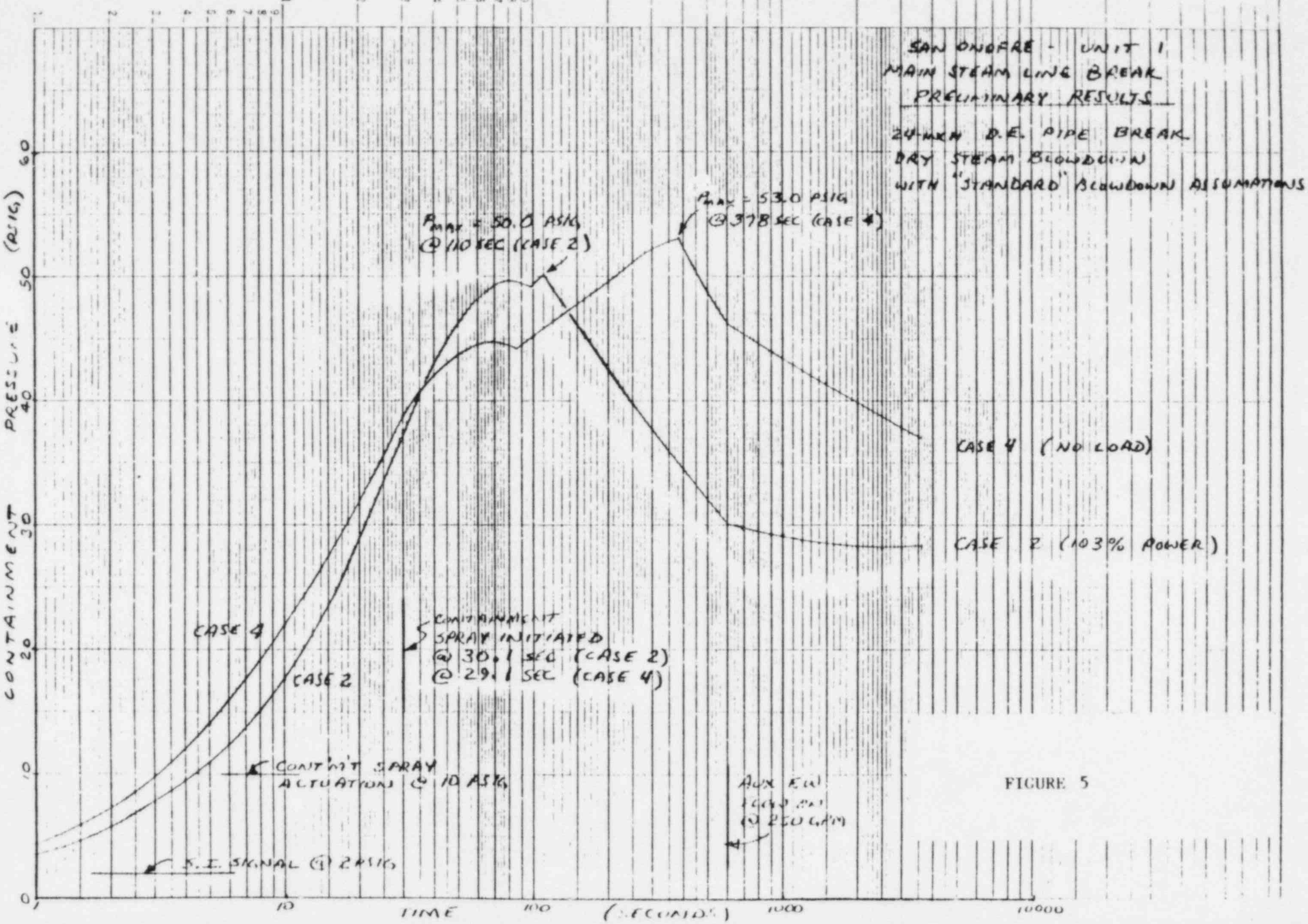

SAN ONOFRE - UNIT 1
MAIN STEAM LINE BREAK
PRELIMINARY RESULTS
24-INCH D.E. PIPE BREAK
DRY STEAM BLOWDOWN
WITH "STANDARD" BLOWDOWN ASSUMPTIONS
P_{max} = 53.0 PSIG
@ 378 SEC (CASE 4)
P_{max} = 50.0 PSIG
@ 110 SEC (CASE 2)
CASE 4 (NO LOAD)
CASE 2 (103% POWER)
CASE 4
CASE 2
CONTAINMENT
SPRAY INITIATED
@ 30.1 SEC (CASE 2)
@ 29.1 SEC (CASE 4)
CONT'M'T SPRAY
ACTUATION @ 10 PSIG
S.I. SIGNAL @ 2 PSIG
CONTAINMENT PRESSURE (PSIG)
TIME (SECONDS)
0
10
20
30
40
50
60
1
10
100
1000
10000

FIGURE 5

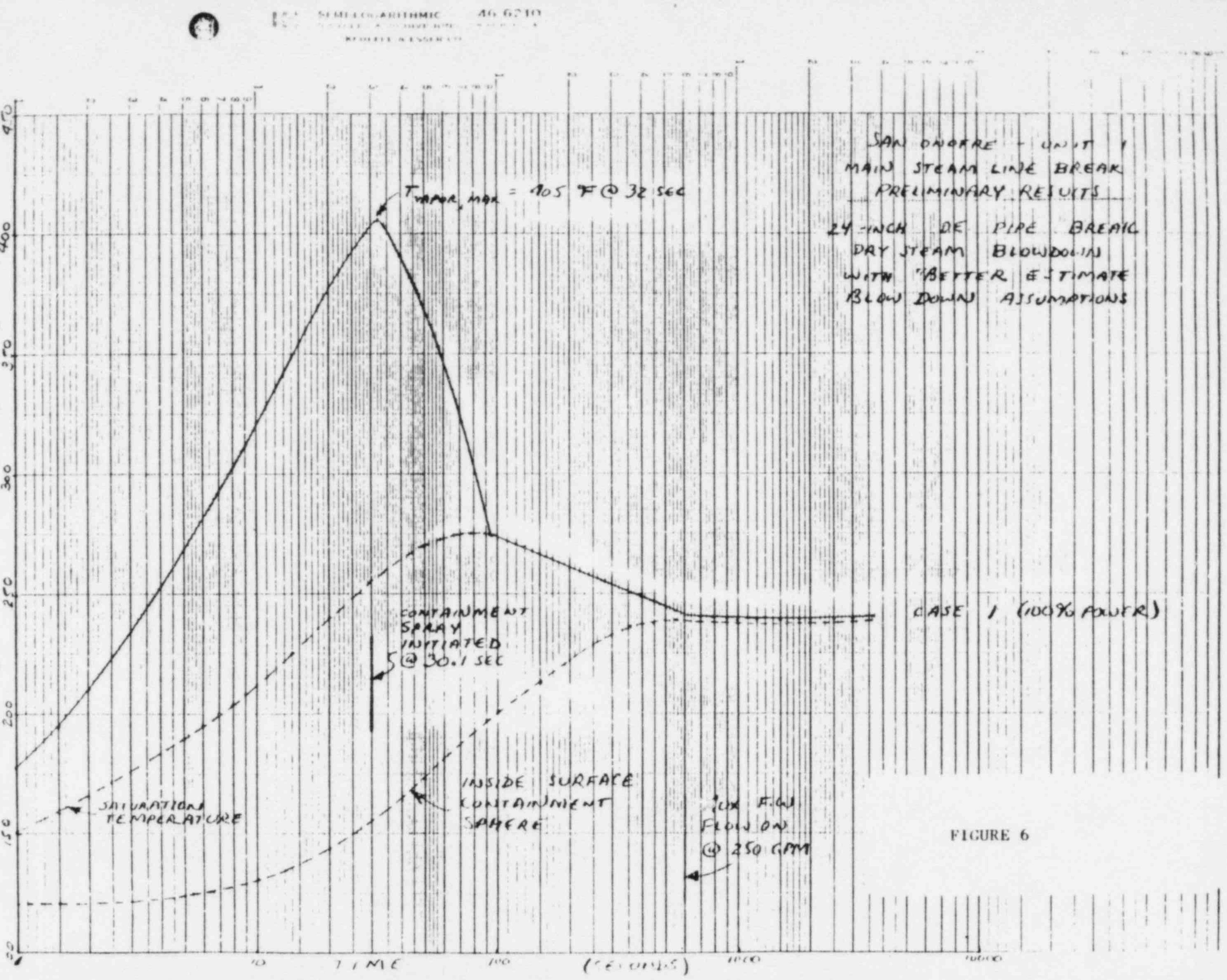
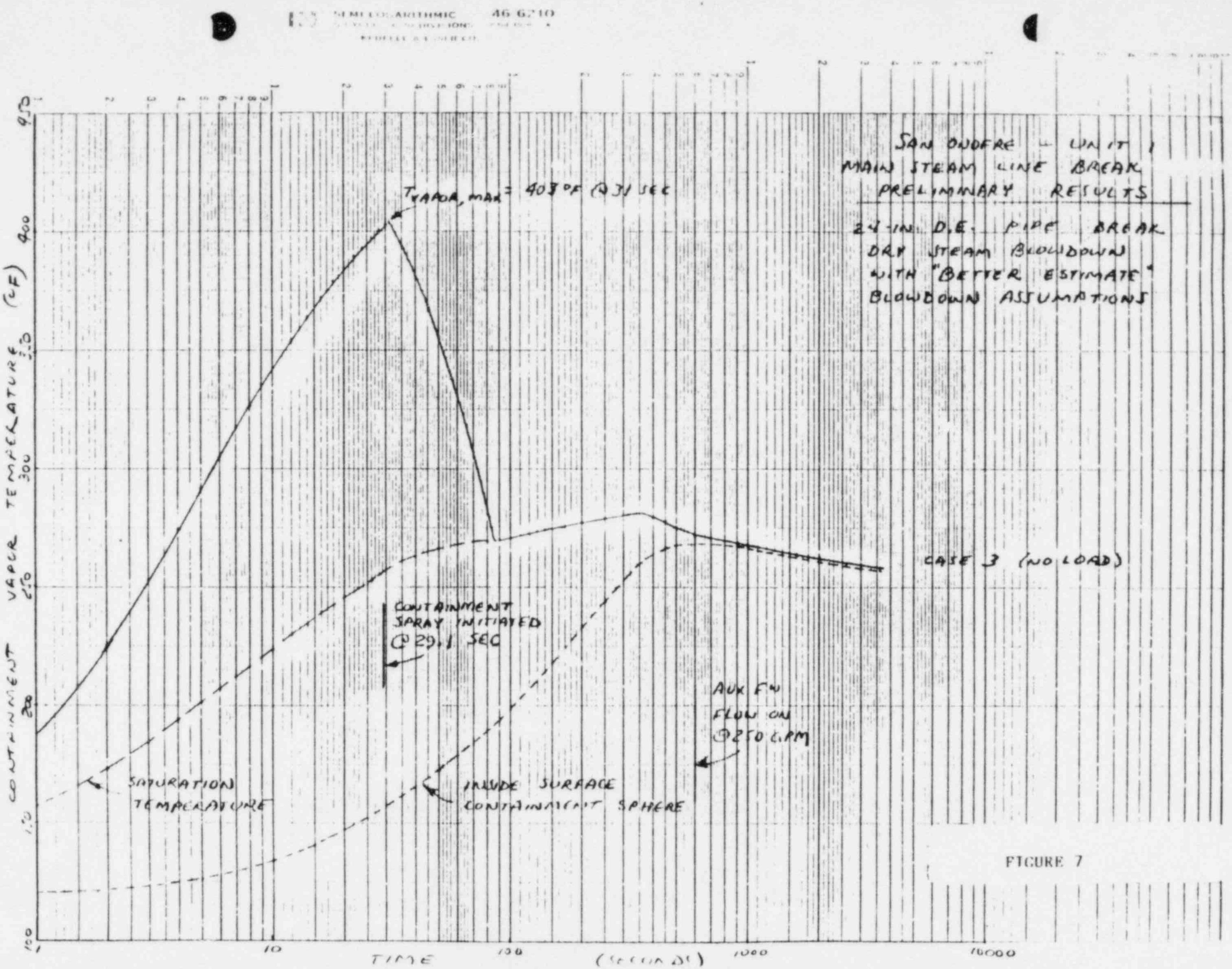

SEMILOGARITHMIC 46 6210
SAN ONOFRE - UNIT 1
MAIN STEAM LINE BREAK
PRELIMINARY RESULTS
24-INCH DE PIPE BREAK
DRY STEAM BLOWDOWN
WITH "BETTER ESTIMATE
BLOW DOWN ASSUMPTIONS
T VAPOR, MAX = 405 °F @ 32 SEC
CONTAINMENT SPRAY INITIATED @ 30.1 SEC
CASE 1 (100% POWER)
INSIDE SURFACE CONTAINMENT SPHERE
SATURATION TEMPERATURE
AUX F.W. FLOW ON @ 250 GPM
CONTAINMENT VAPOR TEMPERATURE (°F)
TIME (SECONDS)
450
400
350
300
250
200
150
100
1
10
100
1000
10000

FIGURE 6



SEMI-LOGARITHMIC 46 6210
SAN ONOFRE - UNIT 1
MAIN STEAM LINE BREAK
PRELIMINARY RESULTS
24-IN. D.E. PIPE BREAK
DRY STEAM BLOWDOWN
WITH "BETTER ESTIMATE"
BLOWDOWN ASSUMPTIONS
T VAPOR, MAX = 403°F @ 31 SEC
CONTAINMENT SPRAY INITIATED @ 29.1 SEC
CASE 3 (NO LOAD)
AUX FW FLOW ON @ 250 GPM
SATURATION TEMPERATURE
INSIDE SURFACE CONTAINMENT SPHERE
CONTAINMENT VAPOR TEMPERATURE (°F)
TIME (SECONDS)
450
400
300
200
100
1
10
100
1000
10000

FIGURE 7

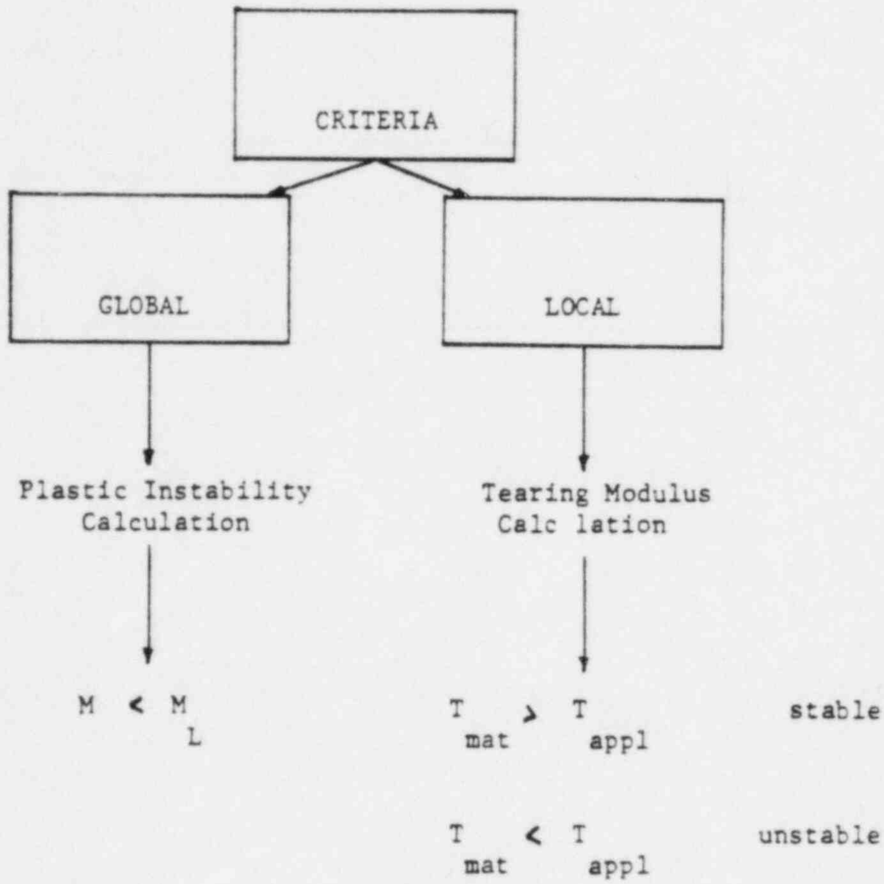

CRITERIA
GLOBAL
LOCAL
Plastic Instability Calculation
Tearing Modulus Calc lation
$M < M_L$
$T_{mat} > T_{appl}$ stable
$T_{mat} < T_{appl}$ unstable

Figure 8

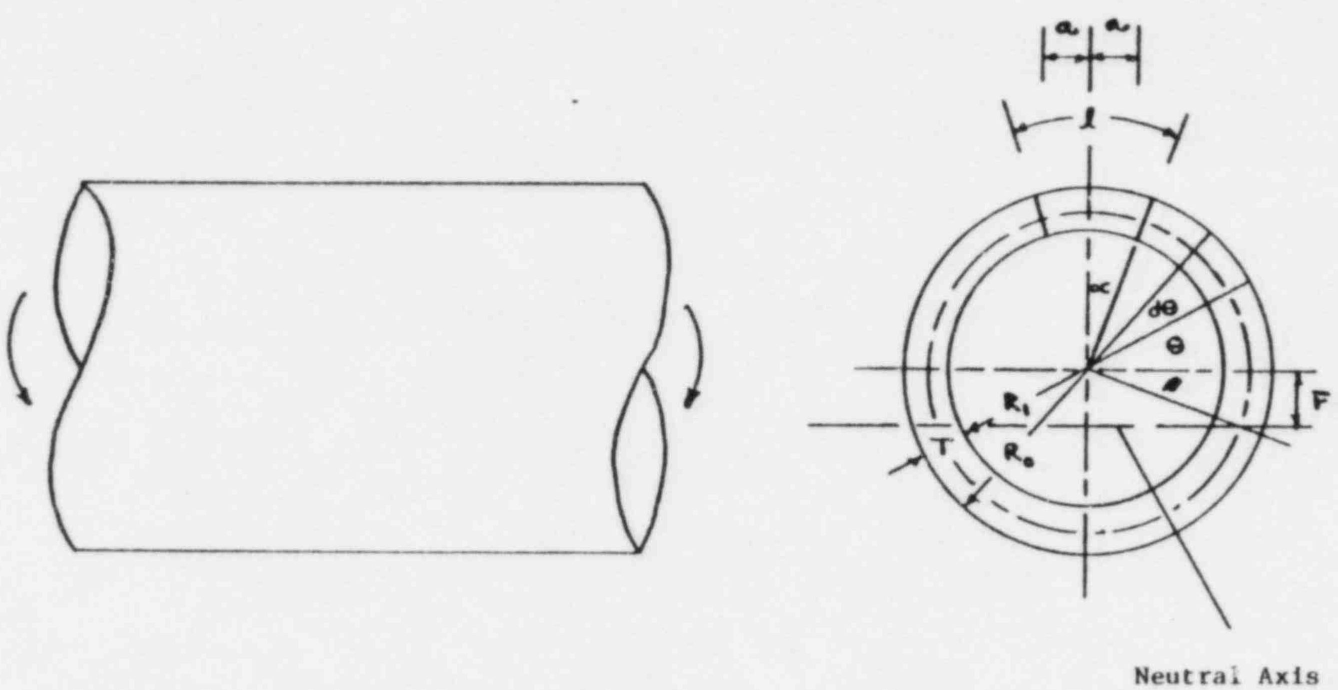


Figure 9- Pipe With a Through-Wall Crack in Bending

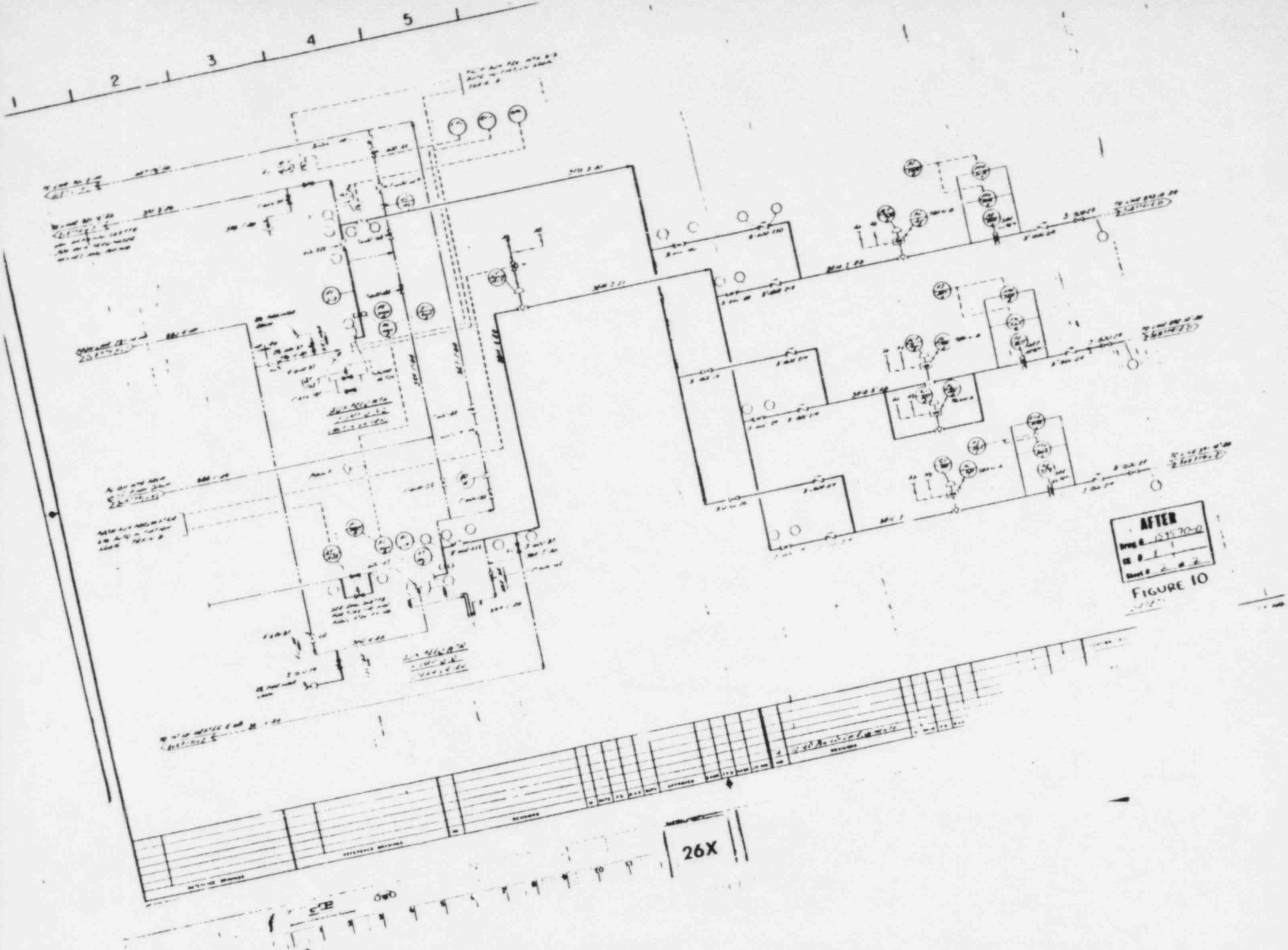
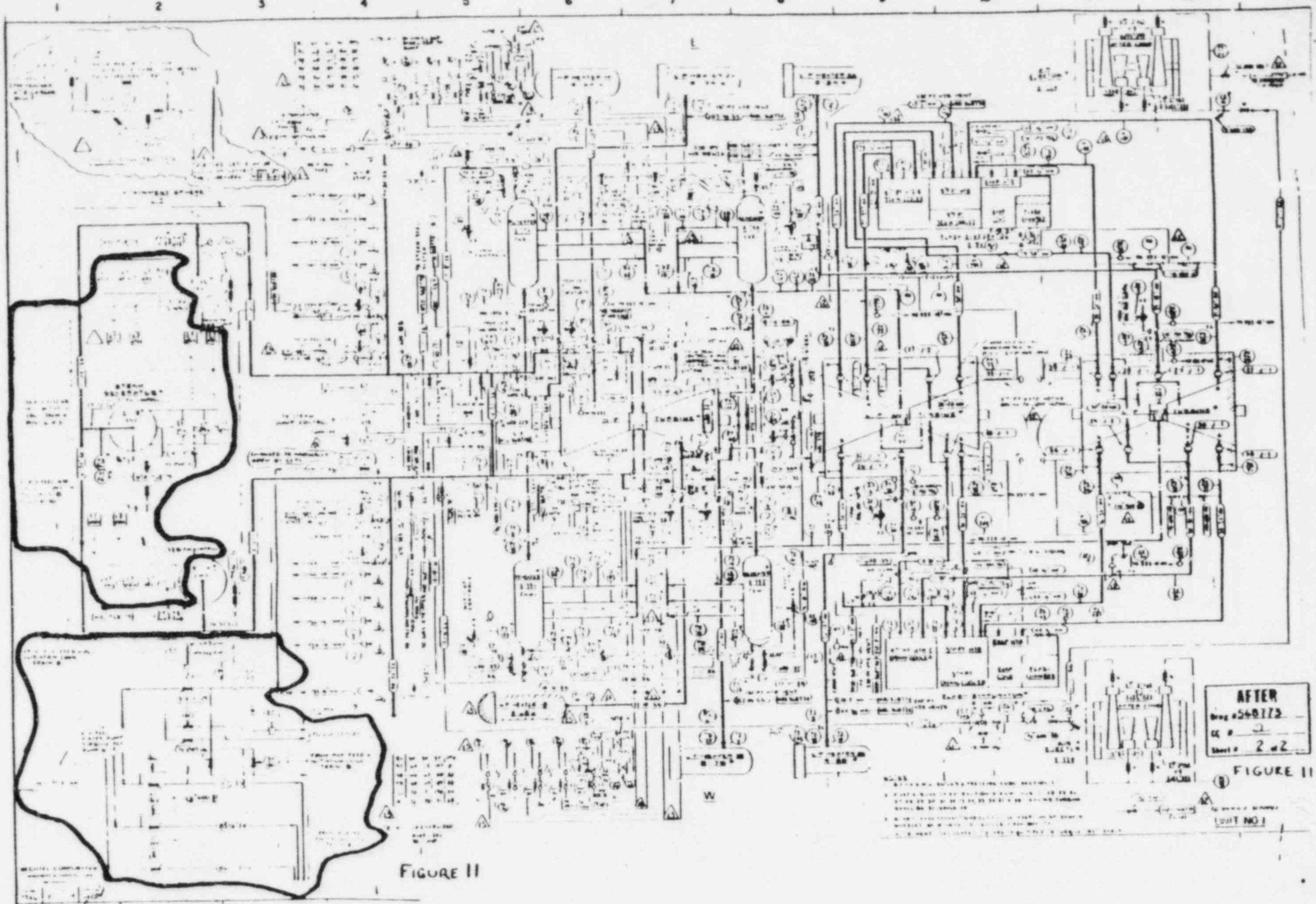


FIGURE 10



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~~Crutchfield~~ 116
Circulate ORB-5
Return to DMC

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 15 1980

MEMORANDUM FOR: P. Check, Assistant Director for
Plant Systems
Division of Systems Integration

FROM: Thomas M. Novak, Assistant Director for
Operating Reactors
Division of Licensing

SUBJECT: AUTOMATIC INITIATION OF AUXILIARY FEEDWATER SYSTEM (AFWS)

Enclosed is Bob Clark's recommendations regarding a priority listing for reviewing the subject matter. I have reviewed the recommendations and concur.

Tm Novak

Thomas M. Novak, Assistant Director for
Operating Reactors
Division of Licensing

Enclosure:
As stated

cc: D. Eisenhut
D. Ross
Operating Reactor BC's
R. Satterfield
G. Lainas

8010220246 2pp.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 11 1980

MEMORANDUM FOR: T. Novak, Assistant Director for Operating Reactors, DL

FROM: R. Clark, Chief, Operating Reactors Branch No. 3, DL

SUBJECT: AUTOMATIC INITIATION OF AUXILIARY FEEDWATER SYSTEM (AFWS)

Mr. P. Check's memorandum to you dated May 13, 1980 on this subject, requested that a priority listing be established for the review of this matter (Lessons Learned Item 2.1.7.a) for operating plants.

We have developed such a listing for the operating PWRs which classifies them into one of three priority categories, "A" being the highest priority. "A" category plants plan to be shutdown for significant periods (2 to 6 weeks) for various reasons (SG inspections, refueling, L^2 implementation, etc.) sometime in the months of August or September. Therefore, these plants should receive the highest priority since they offer the opportunity to implement any hardware fixes required by 2.1.7.a resolution while they are already down for the other purposes mentioned above. Licensees for these plants have submitted sufficient information to permit a review of their proposed implementation.

Category B plants plan to be shutdown in the months of October or November. Category C plants plan to be shutdown in December or later than the January 1, 1981 deadline.

Immediate attention should be paid to the plants to be shutdown in the July August period if the review is to be completed prior to their currently scheduled startup (Robinson 2, Millstone 2 and San Onofre 1).

It should also be noticed that for the multi-unit sites, if the designs are highly similar for all the plants on a site one review may have the effect of resolving the issue for more than one reactor (Surry 1 & 2, Turkey Point 3 & 4, Prairie Island 1 & 2, Oconee 1, 2 and 3, etc.).

The most rapid way to kick this review off appears to be for the assigned ICSB reviewer to promptly visit the PM to assess the quality of the licensee's response to Mr. Denton's letter of October 30, 1979. Technical responses to each of the seven items in that letter may have been

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documented in licensee correspondence in response to the B&O review of AFWS in which case a bit of searching may be required to locate the information. Drawings (P&IDs, electrical schematics and logic diagrams) may no longer be available from the original submittals except in the Docket Files in the basement of the Phillips building. Following this assessment ICSB will know what additional material has to be reproduced as requested from the licensee to permit the review to be done.

The PMs for each plant are listed in Enclosure 3. The scheduled dates for completion of the staff's review and implementation of any required hardware fixes are indicated by Enclosure 2 which provides the expected date of shutdown and duration of shutdown as precisely as these can be determined at this time.

R. A. Clark

R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

CATEGORY A

Plant	PM	Room/Phone	Planned Shutdown Period
Robinson 2	Neighbors	318/27037	August
Salem 1	Ross	318/27134	September
Surry 1	Neighbors	318/27037	September
Turkey Point 3	Grotenhuis	319/27128	August
Millstone 2	Conner	334/27435	August
Prairie Island 1	Martin	P-1122/28087	September
Oconee 1	Fairtile	330B/27435	September
San Onofre	Wambach	322/27038	August

Special

Zion 1 & 2, Indian Point 3 and TMI-1 are in a special review category as already determined by DL and DSI.

ORB#3 (cont'd)	Shutdown Scheduled Before January 1981	Submittal Status	Review Priority
North Anna 1	December 1, Refueling	2	C
Point Beach 1	November 7	2	B
2	No	2	C
Prairie Island 1	September, Refueling	2	A
2	January, Refueling	2	C
St. Lucie 1	No	2	C
Trojan	No	2	C
ORB#4			
ANO-1	No	3	C
Crystal River 3	October for L^2 Implementation	3	B
Davis Besse 1	Shutdown Now, Startup August 1	3	C
Oconee 1	September for L^2 Implementation	2	A
2	October for L^2 Implementation	2	B
3	November for L^2 Implementation	2	B
Rancho Seco 1	No	3	C
TMI-1	Hearing to start Oct. 9		
ORB#5			
Ginna	November. Steam Gen. Inspec.	2	B
Haddam Neck	No	2	C
Palisades	No	2	C
San Onofre 1	Now down due to S. Gen. - problems and refueling. Startup not expected before late August.	2	A
Yankee Rowe	Now down due to turbine problems. Startup not expected before Dec. 80.	2	C

Submittal Status

1. Itemized response including P&IDs, Logic drawings and electrical schematics.
2. Itemized response exists but it is not known if the drawings mentioned in 1. above have yet been sent to NRC for these plants.
3. Response is too abbreviated to support a technical review by the staff at this time.

ENCLOSURE 2

ORB#1	Shutdown Scheduled Before January 1981	Submittal Status	Review Priority
Beaver Valley	Has been down since Nov. 79. Short term L^2 currently under review. Currently scheduled startup in August.	3	C
D.C. Cook 1	No	1	C
2	No	1	C
Farley 1	October - Refueling	2	B
*Indian Point 3	No. Implementation will be per an Order for this plant.		
Kewaunee	No.		C
Robinson 2	Now down due to SG leaks. May refuel. Expected to be down most of August.	1	A
Salem 1	September - Refueling	1	A
Surry 1	September - Repair SG	2	A
2	Down now - startup 7/30	2	C
Turkey Point 3	Aug. or Oct. for 2 weeks for SG inspec.	1	A
4	November	1	B
Zion 1 2	Now under review in ICSB. Implementation will be per an Order for these plants.		
ORB#3			
ANO-2	No	3	C
Calvert Cliffs 1	October 15, Refueling, 6 weeks	1	B
2	January 1	1	C
Fort Calhoun	No.	2	C
Maine Yankee	No.		C
Millstone 2	July 26, Refueling, 6 wks.	2	A

ASSIGNMENT LIST
FOR
OPERATING REACTORS
DIVISION OF LICENSING

May 8, 1980

Thomas M. Novak, A/D for Operating Reactors

Operating Reactors Branch #1 - Steven A. Varga, Chief
C. Parrish, Licensing Assistant

DOCKET NO.	FACILITY	PRIMARY	BACKUP
50-334	Beaver Valley 1	Ross	Miner
50-315/316	D. C. Cook 1/2	Miner	Reeves
50-348	Farley 1	Reeves	Licciardo
50-3/237/286	Indian Point 1/2/3	Olshan	Ross
50-305	Kewaunee	Licciardo	Neighbors
50-261	Robinson 2	Neighbors	Miner
50-272	Salem 1	Ross	Grotenhuis
50-280/281	Surry 1/2	Neighbors	Grotenhuis
50-250/251	Turkey Point 3/4	Grotenhuis	Neighbors
50-295/304	Zion 1/2	Reeves	Olshan

Note: D. Wigginton on detail to NFS

Operating Reactors Branch #2 - Thomas A. Ippolito, Chief
S. Norris, Licensing Assistant

DOCKET NO.	FACILITY	PRIMARY	BACKUP
50-259/260/296	Browns Ferry 1/2/3	Clark	Rooney
50-325/324	Brunswick 1/2	Hannon	Bevan
50-298	Cooper	Rooney	Polk
50-249	Dresden 3	Bevan	Hannon
50-331	Duane Arnold	Kevern	Clark
50-333	Fitzpatrick	Polk	Kevern
50-133	Humboldt Bay	Rooney	Clark
50-263	Monticello	Kevern	Bevan
50-220	Nine Mile Point 1	Polk	Kevern
50-293	Pilgrim 1	Hannon	Polk
50-254-265	Quad Cities 1/2	Bevan	Rooney
50-271	Vermont Yankee	Rooney	Hannon

Note: B. Siegel on detail to HFS
T. Alexion (Intern)

Operating Reactors Branch #3 - Robert A. Clark, Chief
P. Kreutzer, Licensing Assistant

DOCKET NO.	FACILITY	PRIMARY	BACKUP
50-368	Arkansas 2	Martin	Sands
50-317/318	Calvert Cliffs 1/2	Conner	Wagner
50-285	Fort Calhoun	Wagner	Conner
50-70	GETR	Nelson	Requa
50-309	Maine Yankee	Nelson	Requa
50-336	Millstone 2	Conner	Wagner
50-338	North Anna 1	Engle	Trammell
50-266/301	Point Beach 1/2	Trammell	Engle
50-282/306	Prairie Island 1/2	Martin	Sands
50-335	St. Lucie 1	Nelson	Requa
50-344	Trojan	Trammell	Engle

Operating Reactors Branch #4 - Robert W. Reid, Chief
M. Duncan, Licensing Assistant for TMI-1
R. Ingram, Licensing Assistant

DOCKET NO.	FACILITY	PRIMARY	BACKUP
Babcock & Wilcox			
50-313	Arkansas 1	Vissing	Garner
50-302	Crystal River 3	Erickson	Fairtile
50-346	Davis-Besse	Garner	Vissing
50-269/270/287	Oconee 1/2/3	Fairtile	Erickson
50-312	Rancho Seco	Garner	Vissing
50-289	Three Mile Island 1	DiIanni	Silver
	Three Mile Island 1 (hearing)	Silver	DiIanni
General Electric			
50-321/366	Hatch 1/2	Verrelli	Fairtile
50-277/278	Peach Bottom 2/3	Verrelli	Fairtile

Gus C. Lainas, A/D for Safety Assessment

Operating Reactors Branch #5 - Dennis M. Crutchfield, Chief
H. Smith, Licensing Assistant

DOCKET NO.	FACILITY	PRIMARY	BACKUP
50-155	Big Rock Point (GE)	Paulson	Shea
50-10/237	Dresden 1/2 (GE)	O'Connor	Nowicki
50-244	Ginna (W)	Nowicki	Wambach
50-213	Haddam Neck (W)	Caruso	Burger
50-409	Lacrosse (AC)	Shea	Caruso
50-245	Millstone 1	Shea	Paulson
50-219	Oyster Creek (GE)	Paulson	Nowicki
50-255	Palisades (CE)	Wambach	O'Connor
50-206	San Onofre 1 (W)	Wambach	Burger
50-29	Yankee Rowe (W)	Burger	Caruso

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Southern California Edison Company

P. O. BOX 800
2244 WALNUT GROVE AVENUE
ROSEMEAD, CALIFORNIA 91770

July 16, 1980

U. S. Nuclear Regulatory Commission
Attention: R. H. Engelken, Director
Office of Inspection and Enforcement
Region V
Suite 202, Walnut Creek Plaza
1990 North California Boulevard
Walnut Creek, California 94596

Gentlemen:

Subject: Docket No. 50-206
IE Bulletin 80-04, Analysis of a PWR
Main Steam Line Break With Continued
Feedwater Addition
San Onofre Nuclear Generating Station
Unit 1

By letter dated May 19, 1980 we advised you that we would complete our review and provide our response to Item 2 of IE Bulletin 80-04 by July 1, 1980. The purpose of this letter is to reschedule the submittal of our response to Item 2 of IE Bulletin 80-04.

As indicated in our letter of May 19, 1980 as part of our review, Westinghouse provided data which is generic in nature. Additional time was needed to review the applicability of this information relative to San Onofre Unit 1. In the course of our review, it was deemed prudent to perform a partial reanalysis of the MSLB core response to confirm the applicability of the generic conclusion to San Onofre Unit 1. This analysis has been completed and is currently being transmitted to us. Following completion of our review of the results and conclusion of this reanalysis, we will provide our response to Item 2 of IE Bulletin 80-04. We estimate that our review will be completed and response will be submitted by August 1, 1980.

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If you have any questions or desire further information, please contact me.

Very truly yours,

H. L. Ottoson
Manager of Nuclear Operations

cc: D. M. Crutchfield (NRR)
NRC Office of Inspection and Enforcement
(Washington, D. C.)

118

Southern California Edison Company

P O BOX 800
2244 WALNUT GROVE AVENUE
ROSEMEAD CALIFORNIA 91770

August 4, 1980

U. S. Nuclear Regulatory Commission
Attention: R. H. Engelken, Director
Office of Inspection and Enforcement
Region V
Suite 202, Walnut Creek Plaza
1990 North California Boulevard
Walnut Creek, California 94596

Gentlemen:

Subject: Docket No. 50-206
IE Bulletin 80-04, Analysis of a PWR
Main Steam Line Break With Continued
Feedwater Addition
San Onofre Nuclear Generating Station
Unit 1

By letter dated July 16, 1980 we advised you that we would complete our review and provide our response to Item 2 of IE Bulletin 80-04 by August 1, 1980. The purpose of this letter is to provide that response.

Item 2 of IE Bulletin 80-04 requested licensees to review analysis of the reactivity increase which results from a main steam line break inside or outside containment to determine if previous analysis considered all potential water sources and if the reactivity increase is greater than previous analysis indicated.

In response to our request, Westinghouse reviewed the previous analysis of core response following a main steam line break for San Onofre Unit 1. The results of the review showed that no main or auxiliary feedwater had been assumed in the previous analysis. Subsequently, Westinghouse performed a reanalysis of this event. The cases reanalyzed were a main steam line break (complete severance of a pipe) outside containment at no load conditions with offsite power available, and an

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accidental depressurization of the main steam system associated with the inadvertent opening of a single steam dump, relief, or safety valve with offsite power available. These cases conservatively assumed main feedwater flow addition until main feedwater isolation on the safety injection signal and auxiliary feedwater runout flow initiated coincident with the event. The results of the reanalysis confirmed that the main steam line break transient results for these cases are very insensitive to continued feedwater addition for San Onofre Unit 1. It is expected that the results for other no load cases previously analyzed and full load cases (previously shown to be less limiting) would also be insensitive to continued feedwater addition based on Westinghouse generic studies.

The first minute of the transient is dominated entirely by the steam flow contribution to primary-secondary heat transfer, which is the forcing function for both the reactivity and thermal-hydraulic transients in the core. The effect of auxiliary feedwater is minimal. The primary side pressure, on which the low pressurizer pressure safety injection signal is based, decays at a slightly faster rate with the addition of auxiliary feedwater. This accelerates the safety injection signal actuation (< .5 second sooner) as well as allowing a slightly greater safety injection flowrate with the faster pressure decay. These two effects compensate for the increased cooldown rate. The overall results are, therefore, negligibly impacted with the addition of auxiliary feedwater flow.

The auxiliary feedwater flow becomes a dominant factor in determining the duration and magnitude of the steam flow transient during later stages in the transient. However, the limiting portion of the transient occurs during the first minute, both due to higher steam flows inherently present early in the transient and due to the introduction of boron to the core via the safety injection system.

Hence, the conclusions documented in the previously submitted main steam line break core response analysis for San Onofre Unit 1 remain valid and applicable.

If you have any questions or desire further information, please contact me.

Very truly yours,

H. L. Ottoson
Manager of Nuclear Operations

cc: [redacted] (NRR)
NRC Office of Inspection and Enforcement
(Washington, D. C.)

119

Southern California Edison Company

P. O. BOX 800
2244 WALNUT GROVE AVENUE
ROSEMEAD, CALIFORNIA 91770

K. P. BASKIN
MANAGER OF NUCLEAR ENGINEERING,
SAFETY, AND LICENSING

TELEPHONE
(213) 572-1401

October 6, 1980

Director, Office of Nuclear Reactor Regulation
Attention: D. M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
Automatic Initiation of Auxiliary Feedwater System
San Onofre Nuclear Generating Station
Unit 1

References:
(1) D. G. Eisenhut letter to J. H. Drake dated November 15, 1979.
(2) D. L. Ziemann letter to J. H. Drake dated December 21, 1979.
(3) K. P. Baskin letter to D. G. Eisenhut dated January 16, 1980.
(4) K. P. Baskin letter to D. L. Ziemann dated April 29, 1980.
(5) IE Bulletin No. 80-04 dated February 8, 1980.
(6) H. L. Ottoson letter to R. H. Engelken dated May 19, 1980.
(7) J. G. Haynes letter to D. M. Crutchfield dated August 8, 1980.

Enclosure 2 of Reference (1) provided an NRC staff request for information regarding the auxiliary feedwater system (AFWS) flow requirements at San Onofre Unit 1. Reference (2) provides an NRC staff request for information regarding the applicability of current analyses of a main steam line break or main feedwater line break assuming early initiation of auxiliary feedwater flow. In Reference (3) it was indicated that in order to supply the requested information, a complete re-analysis of the applicable transients and accidents would have to be performed and it was estimated that the information could be submitted by October 1, 1980. This same schedule was reiterated in Reference (4), and it was indicated that the design details for the automation of the AFWS would be submitted for NRC staff review by October 15, 1980.

Reference (5) was a request from the NRC Office of Inspection and Enforcement for information similar to that requested by Reference (2). In Reference (6) it was indicated that the information requested in Reference (5) would be supplied by October 1, 1980 in conjunction with our submittal to the NRC staff of the results of the main steam line break re-analysis.

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In Reference (7), the NRC staff was provided with information regarding the impact of the failure of the salt water cooling system on the design basis loss of coolant accident and the main steam line break accident. Included in Reference (7) was a committment to advise the NRC staff if the above discussed re-analyses, due October 1, 1980, would impact the information supplied in Reference (7).

The purpose of this letter is to reschedule the date for submittal of the above discussed information which was to be supplied by October 1, 1980. A delay has resulted due to the repair work which is being done on the steam generators at San Onofre Unit 1. As a part of the analytical effort associated with the steam generator repairs, it was determined that the Reload Safety Evaluation for Cycle 8 would be impacted and it was necessary to perform re-analyses of the affected events. Therefore, it is expected that the requested information can be submitted by December 1, 1980. The design details for the automated auxiliary feedwater system will be provided by October 15, 1980 as indicated in Reference (4).

If you have any questions or desire additional information, please contact me.

Very truly yours,

K P Baskin

cc: R. H. Engelken (NRC Office of Inspection and Enforcement, Region V)
Division of Reactor Operations Inspection (NRC Office of Inspection and Enforcement, Washington, D.C.)

AD12
SC1
TMI

October 6, 1980

MR. D. K. NELSON

SUBJECT: Additional TMI Information Requirements
San Onofre Nuclear Generating Station
Unit 1

The NRC staff has requested several items of information regarding the modifications which have resulted from the TMI Lessons Learned requirements. As previously discussed with D. F. Martin by phone, the assistance of the Engineering Disciplines is needed in order to respond to the NRC by the committed date. Each of the three items of information is discussed below.

AUXILIARY FEEDWATER SYSTEM DESIGN DETAILS

By letter dated April 29, 1980, it was indicated that the design details of the automatically initiated auxiliary feedwater system (AFWS), to be installed in January, 1981, would be provided to the NRC staff by October 15, 1980. These design details should include the latest drawings (CCN's included) and a funtional description of the system.

Please provide the requested information by October 10, 1980 to support submittal to the NRC by October 15, 1980.

REACTOR COOLANT SYSTEM VENTING DESIGN DETAILS

By letter dated May 22, 1980 it was indicated that the design details of the reactor coolant system vents would be provided to the NRC staff by October 1, 1980. The NRC staff has been contacted and submittal of this information has been deferred to October 15, 1980. The design details should include the latest drawings (CCN's included) and a functional description of the system.

Please provide the requested information by October 10, 1980 to support submittal to the NRC by October 15, 1980.

AUXILIARY FEEDWATER SYSTEM FLOW ANALYSIS

The letter from R. H. Verbeck dated September 12, 1980 provided flow capabilities of the automated Auxiliary Feedwater System (AFWS) to be installed in January, 1981 under conditions of loss of main feedwater (intact system). In order for Westinghouse to complete analysis to meet NRC information requests, we need to provide AFWS flow capabilities under conditions of main steamline break (MSLB) or main feedline break (MFLB). Accordingly, we have identified the following additional AFWS flow analysis:

1. Case 1 - MSLB outside containment. Determine total flow to SG's assuming motor-driven auxiliary feedwater pump feeding 3 SG's through 4 FCV's at 0 psig SG pressure.

2. Case 2 - MSLB inside containment. Determine total flow to 3 SG's through 4 FCV's at 50 psig SG pressure for:

   a. motor-driven pump only
   b. steam-driven pump only
   c. both pumps operating.

3. Case 3 - MFLB outside containment. Determine total flow to SG's assuming motor-driven auxiliary feedwater pump feeding 2 SG's at 0 psig SG pressure and 1 auxiliary feedwater line spilling to atmosphere at 0 psig assuming the 4 FCV's remain fully open. Repeat assuming auxiliary feedwater is isolated from the break (operator action to close 2 FCV's in spilling auxiliary feedwater line).

To be consistent with the NRC committed submittal date (December 1, 1980) and Westinghouse need dates, please provide the information by October 15, 1980.

If you have any questions or desire additional information, please contact me.

WCM

W. C. MOODY

RO
RO/WGF:wpo

cc: D. F. Martin
A. J. Brough
E. J. Donovan/W. G. Flournoy
R. W. Krieger/R. Ornelas
U. D. Shendrikar
A. T. Kaneko
R. H. Verbeck
K. M. Evans
NE Files
[redacted]

RWK/WGF

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Southern California Edison Company

P. O. BOX 800
2244 WALNUT GROVE AVENUE
ROSEMEAD, CALIFORNIA 91770

K. P. BASKIN
MANAGER OF NUCLEAR ENGINEERING,
SAFETY, AND LICENSING

TELEPHONE
(213) 572-1401

October 16, 1980

Director, Office of Nuclear Reactor Regulation
Attention: D. M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Gentlemen:

Subject: Docket No. 50-206
Design Detail Information
San Onofre Nuclear Generating Station
Unit 1

By letter dated October 6, 1980 we committed to provide the NRC staff with the design details for the automated auxiliary feedwater system which will be installed as part of the TMI Lessons Learned Requirements. By letter dated October 9, 1980 we committed to provide the design details for the reactor coolant system vents, also to be installed as part of TMI Lessons Learned Requirements. Accordingly, the required information is provided in Enclosures 1 and 2.

It should be noted that the information provided in this submittal supersedes the information provided in our letter dated June 10, 1980.

If you have any questions or desire additional information please contact me.

Very truly yours,

K P Baskin

Enclosures

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ENCLOSURE 1
AFWS DESIGN DETAILS
SAN ONOFRE UNIT 1

I. Introduction

The Auxiliary Feedwater System (AFWS) will be modified to achieve automatic initiation of both the turbine and motor driven pumps, remote flow control capability and redundant pump discharge trains. The proposed system is shown on enclosed drawing SK-10-14-80.

II. System Description

The steam driven pump (G-10) will be provided with all automatically operated valves required to start the turbine. A pneumatically operated on-off valve (CV 3201) will be installed downstream of existing pressure reducing valve (CV 113), in line 69-3-EG. The existing manual valve will be retained. Isolation valves are provided to allow maintenance on the pneumatic valve. A orificed continuous drain and an intermittent drain controlled by a solenoid valve (SV 3211) are provided upstream of CV 3201. An orificed bypass controlled by solenoid valve SV 3200 or a manual valve is provided for turbine warm-up. The turbine drains are orificed continuous drains and are routed down to a sump. In parallel with these drains solenoid valves (SV 3202, SV 3203, and SV 3204) are provided in lines routed out of the turbine building. This arrangement allows high rate draining during turbine warm-up and also provides a turbine drain system without water pockets.

A pump cooling water supply is controlled by solenoid valve SV 3205. At the suction of pump G-10, a fire water connection for back-up cooling water is provided.

On the discharge of motor driven pump G-10S, a motor operated discharge valve (MOV 1202) will be installed. On the discharge of the turbine driven pump G-10, a pneumatically operated valve (CV 3203) will be installed. Pump G-10S also has a motor operated valve (MOV 1204) discharging to the main feedwater line downstream of feedwater heater E-6B. Individual discharge lines 397A-4"-EG and 381A-4"-EG are provided from the pumps to the control valve manifold. At the control valve manifold each discharge line splits into three lines. Each of the six lines is provided with an isolation valve and a check valve. The six lines are then recombined into three lines 381A-3"-EG, 381B-3"-EG, and 381C-3"-EG so that both pumps are capable of providing flow to each steam generator. Each of the three lines has a flow element (FE 3453, FE 3454, and FE 3455) to provide flow indication in the control room. Four remotely operated control valves (FCV 3300, FCV 3301, FCV 2300, and FCV 2301) are provided in the three lines. The valves are arranged so that lines 381A-3"-EG and 381C-3"-EG each have one valve, and that line 381B-3"-EG has two valves.

Downstream of the control valves, check valves and isolation valves are provided in each line. The lines are then connected to the three main feedwater lines.

The system design and materials will meet the original system design specifications and code.

III. System Operation

A. Automatic Initiation Logic

The AFWS has the capability to be initiated automatically or remote manually from the Main Control Room (MCR). Local manual control capability of the system is retained. Automatic initiation is based on a low steam generator level signal processed from the newly installed level transmitters associated with each of the steam generators.

In order to avoid any spurious and unwanted actuations, the automatic initiation logic will be based on a two out of three (2/3) low level signal logic. Remote manual initiation is accomplished by manual actuation of a control switch in the MCR. Appropriate indication and annunciation is provided in the MRC when automatic or manual initiation of the AFW system occurs. In addition, remote control capability to operate components of the system from the MCR are provided.

The AFWS consists of independent and redundant pump and valve trains providing AFW flow to the steam generators. The automatic initiation signals and circuits are designed such that a single failure will not result in the loss of AFWS function.

Each of the automatic initiating circuits which are powered from station vital power sources receives independent signals from the level transmitters associated with each of the steam generators. The level transmitters supply signals to the logic rack installed in the MCR area, where a low level signal and a 2/3 logic is developed. A control board in the MCR provides monitoring and manual control capability of the system. When the steam generator low level signal through 2/3 logic is initiated, the operator will be informed through appropriate status indication and annunciation in the MCR. However no operator action is required to initiate flow since the system is automatically initiated. AFW flow to each of the steam generators is controlled remote manually by throttling the AFW flow control valves. Appropriate annunciation is provided alerting the operator that the system has been automatically initiated and that flow to the steam generators should be throttled by the operator. The logic is designed such that once the automatic signal is initiated, actuation will occur and the system will remain in the actuation mode until the system is reset when the steam generators achieve normal level. In case the automatic system fails, a system level manual initiation will achieve the same function. This system level manual and automatic initiation capability is independent of individual component control capability from the MCR.

B. Motor Driven Pump

The motor driven pump G1CS and motor operated control valve MOV-1202 are designed to operate automatically upon receipt of the automatic initiation signal. When the system is in the auto mode, upon receipt of the auxiliary feedwater acutation signal, the pump will start automatically and open the discharge control valve MOV-1202. This introduces the auxiliary feedwater flow into the steam generators through the pre-positioned flow control valves without any operator intervention. The pump and the discharge valves are powered from the emergency buses and are part of simultaneous or sequential loading under postulated conditions. Under a loss of power (LOP) condition, the pump will trip and the valve will fail as is. Upon resumption of power, the pump will automatically start after a 20-second time delay if the demand is present. The valve will remain in its last position and thus, flow will resume as the pump builds speed. Under the condition SIS occurs simultaneously with LOP, the 20-second time delay provides sequential loading of the pump on the emergency buses.

If manual control is required, the operator can select the manual operating mode from the auxiliary feedwater panel. In this mode, the system will remain operating without change. However, the operator can start or stop the pump and/or open or close the MOV 1202 manually.

The pump is provided with a pump suction pressure sensor to trip the pump in the event of low suction pressure. The pump is tripped when in the automatic mode and a low suction pressure signal is present for longer than 20 seconds.

C. Steam Driven Pump

The steam driven pump G1C, turbine control valves, and pump discharge control valves CV-3213 are designed to operate automatically on demand of the auxiliary feedwater initiation signal. When the system is in the auto mode and in receipt of the auxiliary feedwater actuation signal, the following events will occur:

- The lube oil cooler water supply valve is opened to provide cooling water to the pump, simultaneously;

- The main steam drain valves are opened for ten seconds to drain the condensate out of the main steam line. After the line has had ten seconds to drain, the valves are automatically closed, then;

- The main steam bypass valve (mounted in parallel with the main steam valve) is opened to preheat the turbine. After a sufficient preheat period, the main steam valve is opened at a controlled rate.

- Once the main steam valve is opened, the drain valves on the turbine steam chest are closed and the steam turbine is operated at full power under governor control. Positive position indication is provided on all valves to provide valve position on the auxiliary feedwater control panel in the main control room. A position switch is also provided at the steam speed governor control station. This switch will alert the operator in the event that the turbine has tripped on overspeed.

- After the main steam valve is fully opened, the pump discharge valve CV 3213 is opened. This will start the auxiliary feedwater flow to the steam generator through the prepositioned flow control valves.

For the turbine driven pump train the operator can select the manual operating mode from the auxiliary feedwater panel. In this mode, the system will remain operating with no change until the operator takes deliberate action. Once the system has been placed in the manual mode, the operator can start or stop the pump in an automatic time sequence or he can manipulate any of the seven valves manually. The pump is provided with a pump suction pressure sensor to trip the pump in the event of low suction pressure. The pump is tripped when in the automatic mode and a low suction pressure signal is present for longer than 20 seconds.

Additional manual backup is provided locally at the pump. All valves have been provided with manual override or separate manual bypass valves.

The operation of the steam driven pump and associated valve train is independent of offsite or onsite AC power.

IV. Flow Indication and Control

Flow indication and control is included on the auxiliary feedwater panel to provide the operator process feedback information for manual operation of the four (4) remote manual control valves. Two parallel flow transmitters are connected to a single orifice plate for flow measurements. Each of the three headers is provided with flow indication to the steam generators upstream of the flow control valves. This allows the operator to monitor and control the flow to each of the steam generators over a flow range of 30-300 gpm. This flow indication is provided with a backup from steam generator level indication (also on the auxiliary feedwater panel). For train separation, all three flow indicators are on one flow train and level indicators are placed on the redundant train. Thus, in the event that a single train is lost, the status of the steam generators will be provided to the operator on the auxiliary feedwater panel in the main control room area.

The four (4) auxiliary feedwater flow control valves are divided into two redundant trains, thus with a single failure in one train, the other train will control the remaining two flow control valves. This allows flow to continue to at least one of the three steam generators. In addition, the valves can be operated locally (and local steam generator level indication is provided).

V. Alarm Logic

Alarms for both trains are provided on the auxiliary feedwater panel to alert the operators if any components are not in their preset ready position. Additional alarms are provided for the positive position indication of four auxiliary feedwater control valves. If the valves are not in their preset position prior to automatic initiation, annunciation in the main control room will warn the operator of this condition.

VI. Periodic Testing

The automatic system is designed with on line testing capability. In order to avoid unnecessary component operation during periodic testing, the operator is able to test the initiation logic without actuating the pump or valves by means of the auto/manual control switches on the auxiliary feedwater panel. Whenever the system is disabled even for testing, annunciation is provided to indicate that the system is not in automatic initiation mode. If, however, during the testing phase, the automatic system is initiated from real process conditions, the operator will receive an alarm. He can either place the system back to the automatic mode or he can start the system in the manual mode. Flow indicators and steam generator level indicators can be tested by providing simulated signals through the test equipment with the final drive elements disabled.

ENCLOSURE 2
RCS VENTING SYSTEM DESIGN DETAILS
SAN ONOFRE UNIT 1

I. Introduction

The Reactor Coolant System Vents (RCSV) are designed to vent noncondensible gases from the reactor head, hot legs A and B, and the pressurizer. The proposed system is shown on enclosed drawing SK-7-15-80.

II. System Description

The reactor vessel head is provided with redundant sets of vent and block valve combinations. The configuration consists of two parallel block valves (SV 2401 and SV 3402) in series with two parallel vent valves (SV 2401 and SV 3401). The pressurizer venting system similarly consists of two parallel block valves (SV 2404 and SV 3404) in series with two parallel vent valves (SV 2403 and SV 3403). This arrangement provides direction of vented gases from the reactor head and the pressurizer either to the pressurizer relief tank or directly to the containment.

All the valves are solonoid operated with positive position indication provided in the main control room (MCR). Although not required per the lessons learned requirements, each venting location is provided with redundant valves to assure venting when desired as well as to avoid undesired possibility of a valve being stuck open.

Each set of block and vent valves is powered from redundant emergency power buses (vital A.C.). The reactor head venting system valves SV 2402 and SV 3402 and pressurizer venting valves SV 2404 and SV 3404 are powered from vital AC derived from DC bus 1, whereas valves SV 2401 and SV 3401 associated with the reactor head venting and valves SV 2404 and SV 3404 associated with the pressurizer venting are powered from vital AC derived from DC bus 2. This powering arrangement will always assure opening and closing of the vent lines when desired.

The valves are qualified to the latest regulatory and industry standards requirements and their qualification is on file.

As required by NUREG-0578, as clarified, leakage detection must be sufficient to identify the leakage through the vent system. Since the system design includes positive position indication for each vent valve and the leakage path is either to other closed systems or directly to the containment, leakage through the vent systems can be identified as described in Section 3.1.4 of the San Onofre Unit 1 Technical Specifications.

III. System Operation

The RCSV is designed to limit flow to less than 90 gpm. This design allows the venting of approximately 42,000 SCFH. This is the amount of H_2 produced during the first 48 hours following an accident, assuming 17% core metal-water reaction per Westinghouse report WCAP-9636.

To eliminate inadvertent operation, the system design requires that the operator first energize the valve train then separately open each vent and block valve. Thus three separate operations are required to accomplish venting. The operator also has a choice of directing the vented gases either to the pressurizer relief tank or directly to containment. The use of the pressurizer relief tank allows the operator to test the system or make small releases without venting coolant directly to containment. Additionally, since the tank has a rupture disk set at 7 psi, this route can also be used to vent to containment should the other block valve fail. The vent path to containment includes a 10" diameter flash pot to separate liquid and vapor.

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BASKIN,K.P. Southern California Edison Co.
RECIP.NAME RECIPIENT AFFILIATION
CRUTCHFIELD Operating Reactors Branch 5

SUBJECT: Requests extension until 810201 to evaluate impact of automating auxiliary feedwater sys on existing safety analysis & extension until 810115 to provide info re main steam line piping integrity evaluation.

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December 5, 1980

Director, Office of Nuclear Reactor Regulation
Attention: D. M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
NRC Requested Information
San Onofre Nuclear Generating Station
Unit 1

By letter dated October 6, 1980 it was indicated that the NRC staff would be provided by December 1, 1980 with the results of the analysis of the impact of automating the Auxiliary Feedwater System (AFWS) on the existing safety analysis for San Onofre Unit 1. Since the analytical effort required to develop the necessary information is quite extensive, additional time will be required to complete the evaluation. It is estimated that the information will be submitted by February 1, 1981.

By letter dated October 9, 1980 it was indicated that the NRC staff would be provided by December 1, 1980 with additional information regarding the Main Steam Line Piping Integrity Evaluation. We have been informed by Westinghouse that the evaluation would not be available in time to support the previously indicated date. It is estimated that the information will be submitted by January 15, 1981.

The above described schedule changes have been previously discussed with members of your staff. If you have any questions or desire additional information, please contact me.

Very truly yours,

K P Baskin

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January 5, 1981

Director, Office of Nuclear Reactor Regulation
Attention: D. M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
Clarification of TMI Action Plan Requirements
San Onofre Nuclear Generating Station
Unit 1

Your letter dated October 31, 1980, forwarded NUREG-0737 containing all TMI-related items approved for implementation by the Commission at that time. You requested that we provide confirmation that the implementation dates for all approved TMI-related items contained in NUREG-0737 will be met, or propose revised dates with justification for delay and planned compensating safety actions during the interim. By letter dated December 23, 1980, we advised you that a response to your October 31, 1980 letter would be submitted by January 5, 1981.

The purpose of this letter is to advise you that we have completed our review of NUREG-0737, as well as previous correspondence addressing TMI-related items. Based on our review, we intend to meet the implementation dates contained in NUREG-0737, with the following exceptions:

II.B.1	Reactor Coolant System Vents
II.B.2	Design Review of Plant Shielding and Environmental Qualification of Equipment For Spaces/Systems Which May Be Used In Post Accident Operations
II.B.3	Post Accident Sampling Capability
II.F.1,	Attachment 1, Noble Gas Effluent Monitor
II.F.1,	Attachment 2, Sampling and Analysis of Plant Effluents
II.F.1,	Attachment 3, Containment High-Range Radiation Monitor
II.F.1,	Attachment 4, Containment Pressure Monitor
II.F.1,	Attachment 5, Containment Water Level Monitor
II.F.1,	Attachment 6, Containment Hydrogen Monitor

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The implementation of these TMI Action Plan Requirements is targeted for completion by July 1, 1982 (II.B.1) and January 1, 1982 (II.B.2, II.B.3 and II.F.1, Attachments 1 through 6) as required. However, a precise completion date cannot be established u il the current steam generator repair is completed and San Onofre Unit 1 resumes power operation. Following completion of the steam generator repair, it is expected that operation of San Onofre Unit 1 will be restricted to an appropriate interval prior to an interim shutdown to perform an inspection to monitor the effectiveness of the repair. The shutdown date to implement these TMI Action Plan Requirements will be scheduled with consideration of the interim shutdown required to perform the steam generator inspection.

Accordingly, we will advise you of the precise shutdown date to implement these TMI Action Plan Requirements prior to resumption of power operation following completion of the steam generator repair. If the shutdown date does not permit implementation of these TMI Action Plan Requirements by July 1, 1982 or January 1, 1982, respectively, we will also advise you of any planned compensating safety actions which will be implemented during the interim.

II.E.1.1 Auxiliary Feedwater System Evaluation

By letter dated November 15, 1979, the NRC identified short-term and long-term recommendations to upgrade the Auxiliary Feedwater System (AFWS). All short-term recommendations will be completed by July 1, 1981 as required. However, as stated in the NRC November 15, 1979 letter, the long-term recommendations for improving AFWS reliability will not be fully established until after the completion of related Systematic Evaluation Program (SEP) review topics with regard to internally and externally generated missiles, pipe whip and jet impingement (including main steam and main feedwater breaks inside and outside containment), quality and seismic design requirements, and the effects of earthquakes, tornados and floods and design basis evaluations. Accordingly, by letter dated January 23, 1980, we deferred implementaton of the long-term recommendations pending completion of the integrated assessment of potential modifications identified by review of station design and operation in connection with the above SEP topics.

The modifications to the AFWS to meet the requirements set forth in II.E.1.2, Auxiliary Feedwater System Automatic Initiation and Flow Indication of NUREG-0737 will be completed by July 1, 1981, as stated therein. Pending completion of the SEP and the identification of all long-term recommendations by the NRC for improving AFWS reliability, the modifications implemented in accordance with II.E.1.2 of NUREG-0737 will assure that the AFWS will perform its intended function to mitigate the consequences of design basis events as described in the safety analysis report. In addition, upon completion of the modification to the AFWS to meet the requirements of II.E.1.2 of NUREG-0737, we will terminate the stationing of an operator to promptly initiate adequate AFWS flow to the steam generators. The stationing of an operator was initially directed by IE Bulletin No. 79-06A forwarded by NRC letter dated April 14, 1979, for those facilities for which the AFWS is not automated.

II.E.1.2, Part 2, Auxiliary Feedwater System Flowrate Indication

As discussed in our October 16, 1980 letter, the flowrate indication and control system utilizes one narrow-range steam generator level indicator in conjunction with one AFWS flowrate indicator per steam generator. The design concept is to convert the wide-range indicator installed in January, 1980 as part of the controls grade automatic AFWS to the narrow-range scale. The conversion is necessary to improve signal resolution and accuracy for the automatic actuation of the AFWS. In addition, a second, redundant narrow-range indicator will be installed.

As part of the new requirement of NUREG-0737, we will qualify the existing wide-range steam generator level indicator or replace the indicator with a qualified indicator, if necessary. In accordance with the October 24, 1980 Order for Modification of License of San Onofre Unit 1, the qualification will be completed by no later than June 30, 1982.

II.E.4.2 Containment Isolation Dependability

Position (5) requires that we provide and justify, the minimum containment pressure that will be used to initiate containment isolations by January 1, 1981 and be in full compliance by July 1, 1981. We are continuing to review containment pressure history during normal operation and the accuracy of the containment pressure sensor. The results of the review will be used as a basis for arriving at an appropriate minimum pressure setpoint for initiating containment isolation. It is expected that the review will be completed and the results submitted to the NRC by April 1, 1981. Submittal by April 1, 1981 will permit the completion of any necessary corrective actions by July 1, 1981 as required by NUREG-0737.

II.F.2 Instrumentation For Detection Of Inadequate Core Cooling

Our March 25, 1980 letter clarified our position regarding the need for additional instrumentation for inadequate core cooling (i.e., reactor vessel water level system). We stated that existing instrumentation, used in conjunction with procedural guidelines and operator training, is sufficient, and that additional instrumentation to detect inadequate core cooling is not warranted. Therefore, no additional instrumentation is scheduled for installation by January 1, 1982, as required by NUREG-0737.

II.K.3.2 Report On Overall Safety Effect Of Power-Operated Relief Valve Isolation System

As discussed in our June 13, 1980 letter, the Westinghouse Owners Group is in the process of developing a report (including historical valve failure rate data and documentation of actions taken since the TMI event to decrease the probability of a stack-open PORV) to address the NRC concerns. However, due to the time-consuming processing of data gathering, breakdown and evaluation, the report is scheduled for submittal to the NRC on March 1, 1981 rather than by January 1, 1981 as required by NUREG-0737. The report will be used to support a decision on the necessity of incorporating an automatic PORV Isolation System as specified in II.K.3.1 of NUREG-0737.

II.K.3.5 Automatic Trip Of Reactor Coolant Pumps During Loss-Of-Coolant Accident

As discussed in our June 13, 1980 letter, we have installed the automatic RCP trip design, except the final electrical connection. The details of the automatic RCP trip design were provided as part of our August 29, 1979 response to IE Bulletin No. 79-06C concerning this subject. We concluded that automatic RCP trip coincident with safety injection initiation is appropriate for San Onofre Unit 1 to provide assurance that the peak clad temperatures following all LOCA and non-LOCA transients remain within acceptable limits.

As directed by letter dated October 3, 1979 from the Office of Inspection and Enforcement, Region V, we have not made the final electrical connection of the design change pending review and approval by the Office of Nuclear Reactor Regulation. Following review and approval of the design change by the Office of Nuclear Reactor Regulation, we will make the final electrical connection and place the systems in service.

II.K.3.17 Report On Outages Of Emergency Core-Cooling Systems Licensee Report and Proposed Technical Specification Changes

As discussed in our June 13, 1980 letter, station operating, maintenance and test records for the emergency core cooling systems for the last five years are currently being reviewed to determine (1) outage dates and duration of outages, (2) cause of the outage, (3) systems or components involved in the outage, and (4) corrective action taken. Due to the time-consuming process of data gathering, reduction and evaluation documenting the results of our review will be submitted by April 1, 1981, rather than by January 1, 1981 as required by NUREG-0737. The report will include proposed changes, if determined to be appropriate, to improve the availability of the emergency core cooling systems.

San Onofre Unit 1 is currently shutdown for steam generator repair with the emergency core cooling systems not required to be operable. It is expected that any proposed changes to the testing and maintenance programs determined to be appropriate to improve the availability of the emergency core cooling systems will be implemented prior to resumption of power operation following the steam generator repair.

III.D.3.4 Control Room Habitability Requirements

A preliminary evaluation of control room habitability using the guidelines contained in NUREG-0737 has been completed. Additional time is required to finalize the information contained in the evaluation and identify any modifications shown to be necessary. It is expected that our evaluation will be completed and the report submitted to the NRC by April 1, 1981 rather than by January 1, 1981, as required by NUREG-0737.

As discussed in our June 13, 1980 letter, we will initiate preliminary design and engineering efforts required to implement any modifications shown to be necessary. However, we do not plan to initiate any procurement or construction activities until after the Regulatory Staff has reviewed our evaluations and concurs with them. It is our intention to target the modifications for completion by January 1, 1983. A more precise implementation schedule will be included with the report submitted by April 1, 1981.

If you have any questions or desire further information concerning our commitments discussed above, please contact me.

Subscribed on this 5th day of January, 1981.

Very truly yours,
SOUTHERN CALIFORNIA EDISON COMPANY

By K P Baskin
K. P. Baskin
Manager of Nuclear Engineering, Safety, and Licensing

Subscribed and Sworn to before me on this 5th day of January, 1981

Agnes Crabtree
Notary Public in and for the County of Los Angeles, State of California

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BASKIN,K.P. Southern California Edison Co.
RECIP.NAME RECIPIENT AFFILIATION
CRUTCHFIELD,D. Operating Reactors Branch 5

SUBJECT: Forwards Westinghouse WCAP-9832 & WCAP-9808 re main steam line integrity evaluation.Crack appearing instantaneously would be stable both globally & locally.Initial circumferential flaw would show negligible growth.

SEE Repts. #8101160473

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K. P. BASKIN
MANAGER OF NUCLEAR ENGINEERING, SAFETY, AND LICENSING

TELEPHONE
(213) 572-1401

January 14, 1981

Director, Office of Nuclear Reactor Regulation
Attention: D. M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
Main Steam Line Piping Integrity Evaluation
San Onofre Nuclear Generating Station
Unit 1

By letter dated October 9, 1980, it was committed that the NRC staff would be provided by January 15, 1981 with the final results of the Main Steam Line Integrity Evaluation to complete the review of this subject which was initiated during a meeting with the Regulatory staff held on May 13, 1980, and documented in our letter dated June 10, 1980. Accordingly, two Westinghouse reports are enclosed: (1) WCAP-9832, "Mechanistic Fracture Evaluation of San Onofre Unit 1 Main Steam Line Pipe Containing a Postulated Through-Wall Crack," dated November, 1980, (2) WCAP-9808, "Fatigue Crack Growth Evaluation for San Onofre Unit 1 Main Steam Line Pipe," dated October, 1980.

The objective of the analysis performed in WCAP-9832 is to examine mechanistically whether a crack which is assumed to appear instantaneously, in the main steam line, would become unstable and lead to a circumferential break when subjected to the worst possible combination of plant loadings. The results of the analysis indicate that the crack would be stable both globally and locally so that assuming the worst loading combination, the postulated flaw will not propagate around the circumference of the pipe and cause a guillotine break.

The fatigue crack growth analysis presented in WCAP-9808 was conducted as suggested by Section XI Appendix A of the ASME Boiler and Pressure Vessel Code. The analysis procedure involves postulating an initial circumferential flaw and predicting the growth of that flaw due to an imposed series of stress transients. The results of the analysis indicate that growth of the postulated flaw by fatigue is negligible.

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Based on the information summarized above, in conjunction with the main steam line Inservice Inspection and leak detection capabilities as described in the enclosure to our letter dated June 10, 1980, it is concluded that there is reasonable assurance that a double-ended guillotine rupture of the main steam line is not credible. The additional considerations associated with the review of the Main Steam Line Break Accident will be provided in our submittal of the impact on the safety analysis of the automation of the auxiliary feedwater system as required by the TMI Lessons Learned Requirements. The submittal will be provided to the NRC by February 1, 1981, as indicated in our letter dated October 9, 1980.

If you have any questions or desire additional information, please contact me.

Very truly yours,

KPB/LL

Enclosures

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WCAP 9832

MECHANISTIC FRACTURE EVALUATION OF SAN ONOFRE UNIT 1 MAIN STEAM LINE PIPE CONTAINING A POSTULATED THROUGH-WALL CRACK

By

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C. Y. Yang

November, 1980

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INTEGRITY EVALUATION OF SAN ONOFRE UNIT 1
MAIN STEAM LINE PIPE

1. INTRODUCTION

1.1 Background

Presently, the Main Steam Line Break (MSLB) evaluation of the Pressurized Water Reactor (PWR) system is carried out by postulating non-mechanistic circumferential (guillotine) breaks in which the pipe is assumed to rupture along the full circumference of the pipe. This can result in overly conservative steam pressure loading in the containment. It is, therefore, highly desirable to be realistic in the postulation of main steam line breaks. Presented in this report is the result of an anlaytical study carried out toward establishing that a non-mechanistic type break will not occur within the main steam line and, therefore, possibility of containment structure overpressurization will be precluded.

1.2 Scope and Objective

The general purpose of this investigation is to show that a circumferential flaw which is larger than any flaw that would be present in the main steam line is stable under the worst combination of plant loadings. The fracture criteria proposed for the analysis will examine the local and global stability. The global analysis is carried out by performing a static elastic-plastic finite element analysis of a straight piece of the main steam line pipe containing a circumferential flaw and subjected to internal pressure and external loading. ADINA (1-1) computer code is used for the ~~finite~~ finite element analysis. The elastic-plastic finite element analysis results are used to obtain an estimate for the J integral, which is required for the local stability evaluation.

2. INITIAL FLAW

It is well known that initial flaw* geometry is one of the three pieces of fundamental data needed for a fracture mechanics evaluation of a given component. The other two data are stress field and material properties. Conceivably, the initial flaw geometry to be assumed in a fracture mechanics evaluation of a component would depend on several factors, namely, fabrication, examination testing and inspection. One of the rational means of establishing an initial flaw geometry is from the knowledge of the probability of missing (or detecting) a given size flaw.

Figure 2-1 shows schematically how one would find an initial flaw size, given the probability of missing (or detection of) a given size flaw. The probability of missing a very small flaw will be nearly unity whereas the probability of missing a through-wall flaw will be nearly zero. Contrarily the probability of detection of a very small flaw and a through wall flaw would be nearly zero and unity, respectively. However, no data quantifying these probabilities is yet available for main steam line piping.

Although examination and inspection experiences do not tell us anything about the size of flaws that have been missed, these experiences do provide some qualitative ideas about the sizes. The ASME Boiler and Pressure Vessel Code (BPVC) Section XI (2-1) specifies that flaws longer than 1/4 inch and deeper than 9 percent of the pipe wall shall be repaired during preservice examination. Similarly, during inservice inspection, Section XI requires that flaws longer than 0.55 inch and deeper than 11 percent of the pipe wall shall be repaired.

It has been shown in reference 2-2 that if one assumes that the largest initial crack is a semielliptical flaw of length 2-1/4 inch and 3/8 inch depth, the growth of the crack will be very small for the 40 year design life of the plant.

In this analysis a through wall circumferential flaw of 10 inch length is used conservatively (Figure 2.2).

*Flaw and crack are used interchangeably and mean the same in fracture mechanics evaluation.

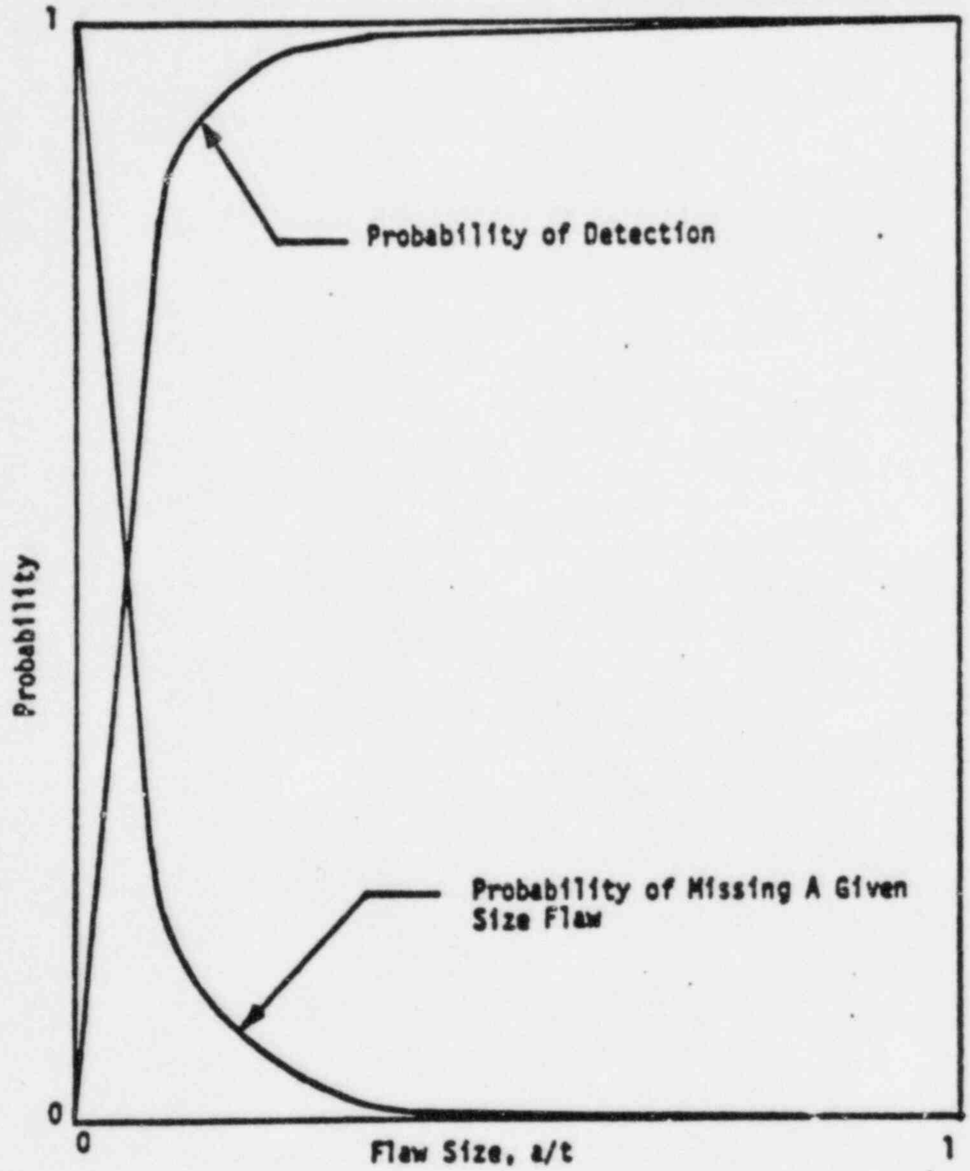

Figure 2.1 Schematic Illustration of Probability Associated with Detecting or Missing a Given Size Flaw.

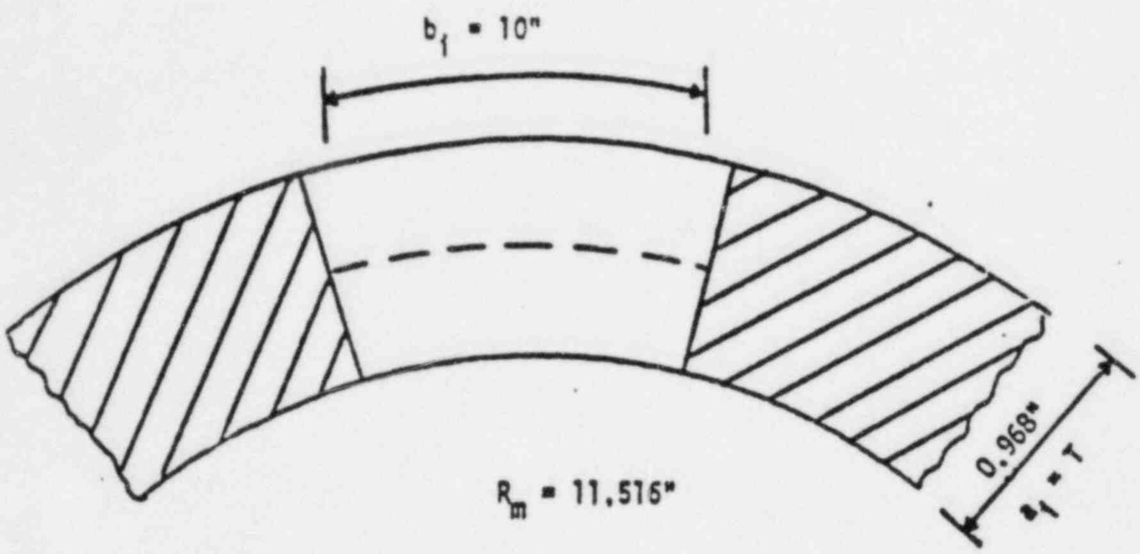

Figure 2.2 - Flaw Shape and Size

3. MATERIAL PROPERTIES

The MSL pipe is made of SA106 Gr B carbon steel material and the size is 24 inch schedule 60. The material properties of SA106 Grade B were obtained from ASME Section III (3-1). Table 3-1 lists those properties. In order to perform the elastic plastic analysis a bilinear stress strain curve as shown in Figure 3-1 was used. (3-1, 3-2).

TABLE 3-1

MATERIAL PROPERTIES

Property (500°F)	Material SA106B
Young's Modulus (psi)	26.4×10^6
Poisson's Ratio	0.3
Yield Point (psi)	28300
Strain hardening Modulus (psi)	0.268×10^6

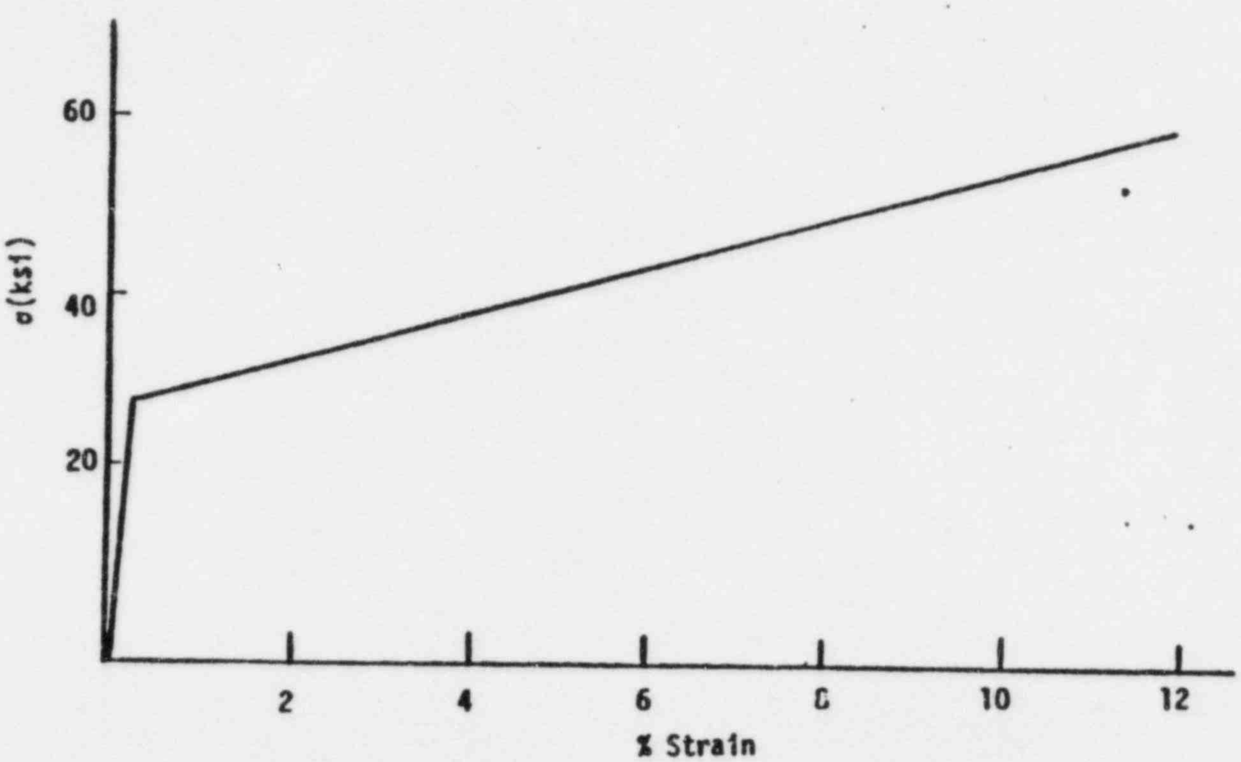

Figure 3-1 Bilinear Stress Strain Curve

4. FAILURE CRITERIA FOR FLAWED PIPES

4.1 General Considerations

Linear elastic fracture mechanics has been accepted as a basis for establishing the fracture capacity of structures made of high-strength low-toughness materials. Active research is being carried on in industry, universities as well as other research organizations to establish fracture criteria for ductile materials. Criteria, being investigated, include those based on J integral initiation toughness, equivalent energy, crack opening displacement, crack opening stretch, crack opening angle, net-section yield, tearing modulus and void nucleation. Several of these criteria are discussed in a recent ASTM publication [4-1].

A practical approach based on the ability to obtain material properties and to make calculations using the available tools, was used in selecting the criteria for this investigation. The ultimate objective is to show that the secondary pipe containing a conservatively assumed circumferential through-wall flaw is stable under the worst combination of postulated and operating condition loads within acceptable engineering accuracy. With this viewpoint, two mechanisms of failure, namely, local and global failure mechanisms should be considered.

4.2 Global Failure Mechanism

For a tough ductile material if one assumes that the material is notch insensitive, then the global failure will be governed by plastic load. Extensive literature is available on this subject. The recent PVRC study [4-2], in critically reviewing the literature as well as data from several hundred tests on pressure vessel heads, nozzles, pipes, elbows and tees, discusses the details of analytical methods, assumptions and methods of correlating experiments and analysis.

A schematic description of the plastic behavior and the definition of plastic load is shown in Figure 4.1. For a given geometry and loading, the plastic load is defined to be the peak load reached in a generalized load versus displacement plot and corresponds to the point of instability.

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A simplified version of this criterion, namely, net section yield criterion has been successfully used in the prediction of the load carrying capacity of pipes containing gross size through-wall flaws [4-2] and was found to correlate well with experiment. This criterion can be summarized by the following relationship:

$$W_a < W_p \qquad (4\text{-}1)$$

where W_a = applied generalized load

W_p = calculated generalized plastic load

In this report, Wp will be obtained by an elastic-plastic finite element analysis of the pipe containing a given size flaw. For a pipe with high $\frac{d}{t}$ ratio and ductile material, the global failure will be the governing mechanism of failure (4-2). For the size of initial flaw proposed in section 2, it is expected that the global plastic load will give a more realistic estimate of the ultimate strength than that provided by the local criteria (i.e. J integral) based loads.

4.3 Local Failure Mechanism

The local mechanism of failure is primarily dominated by the crack tip behavior in terms of crack-tip blunting, initiation, extension and finally crack instability Depending on the material properties and geometry of the pipe, flaw size, shape and loading, the local failure mechanisms may or may not govern the ultimate failure.

The stability will be assumed if the crack does not initiate at all. It has been accepted that the initiation toughness, measured in terms of J_{IN} from a J-integral resistance curve is a material parameter defining the crack initiation. If, for a given load, the calculated J-integral value is shown to be less than J_{IN} of the material, then the crack will not initiate.

If the initiation criterion is not met, one can calculate the tearing modulus as defined by the following relation.

$$T_{app} = \frac{dJ}{da} \quad \frac{E}{\sigma_f^2} \tag{4-2}$$

where T_{app} = applied tearing modulus

E = modulus of elasticity

σ_f = flow stress = $(\sigma_y + \sigma_u)/2$

a = crack length

σ_y, σ_u = yield and ultimate strength of the material, respectively.

In summary, the local crack stability will be established by the two step criteria:

$$J < J_{IN} \tag{4-3}$$

$$T_{app} < T_{mat}, \; J \geq J_{IN} \tag{4-4}$$

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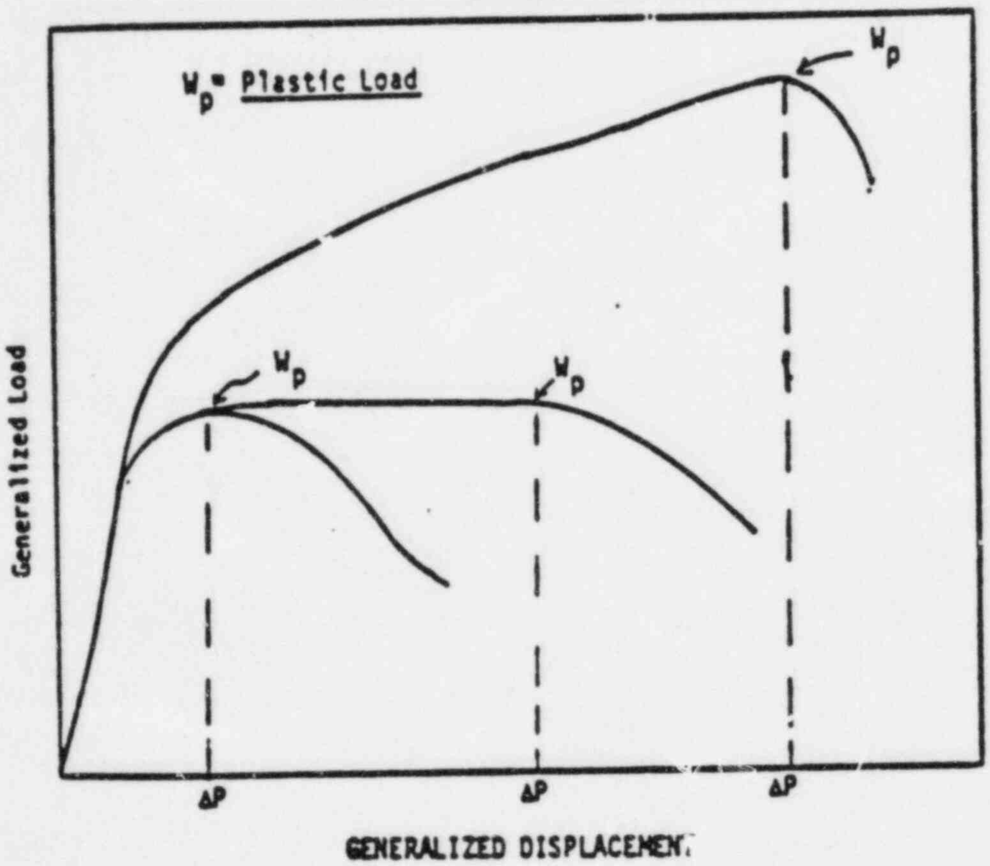


FIGURE 4.1 Typical Load - Deformation Behavior

5. FINITE ELEMENT ANALYSIS OF MAIN STEAM LINE PIPE USING ADINA

5.1 Static Analysis of Precracked Pipe

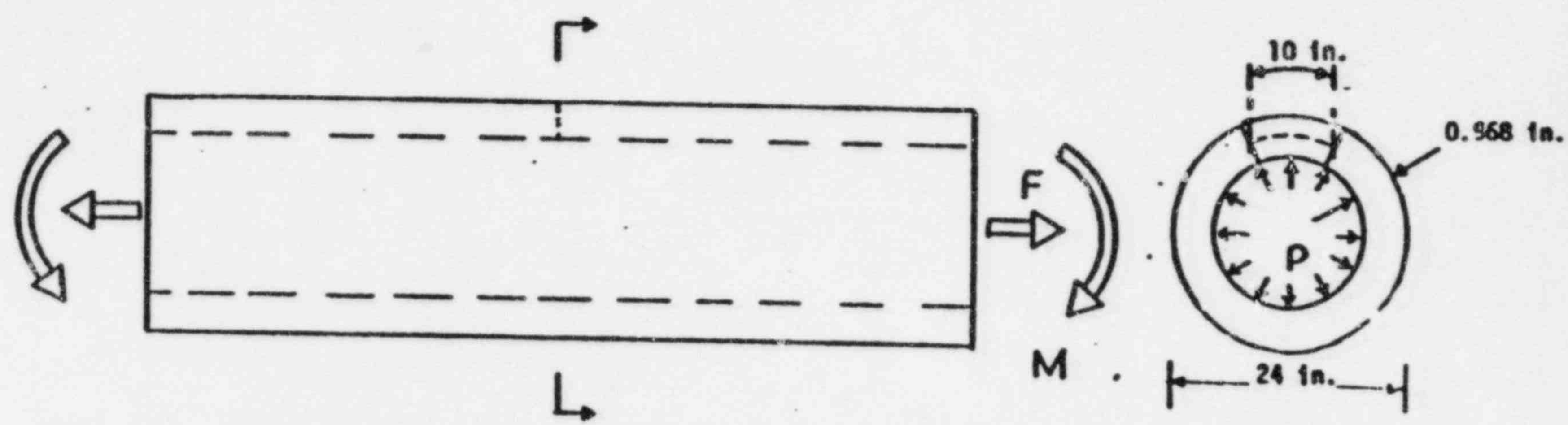
The objective of the finite element analysis is to compute the response to applied load on the main steam line pipe. The geometry of the pipe and the loadings are shown in Figure 5-1. The loadings consist of internal pressure, external bending moment and axial force due to internal pressure acting on the closed end of the pipe.

The length of the main steam line was chosen to be 187 inches. A circumferential through wall 10 inches long crack was postulated and used in the model. Taking advantage of the symmetry, one quarter of the pipe was modeled. Three dimensional variable node isoparametric shell elements were used to model the pipe. Elements were defined by the mid surface node specification. Eight node elements were specified in the vicinty of the crack and four node elements were used away from the crack. Five node (mixed) elements were used for transition. Figure 5-2 shows the finite element model used in the analysis. Figure 5-3 shows the area in the vicinity of the crack.

The material representing the model pipe was assumed to obey von Mises' yield condition and isotropic hardening law. Values of 26.4×10^3 ksi, 2.68×10^2 ksi and 28.3 ksi were used, respectively, for elastic modulus, strain hardening modulus and yield strength.

In performing the elastic-plastic finite element analyses the steam pressure of 710 psia[5-1] and the associated axial loads were applied in 4 equal steps. An external bending moment of 10580 in-kips was then superimposed in 7 equal steps while the pressure was maintained constant. The stiffness was reformulated at every 3rd loading step.

The maximum external moment load of 10580 in-kips used in ADINA calculations is a factor of 3.6 greater than the maximum applied moment of 2880 in-kips[5-2] on the main steam line pipe. This applied moment includes the thermal expansion moment, moment due to dead weight, design basis earthquake and turbine valve closure. Figure 5-4 shows the variation of pipe end slope with increasing pipe moment. It is notable that the slope of the curve is postive at the applied load level: This shows that the cracked pipe is stable under this loading.



P = 710 PSI

F = 0.27×10^6 lbs

M = 10.58×10^6 in-lbs

FIGURE 5-1 MAIN STEAM LINE PIPE

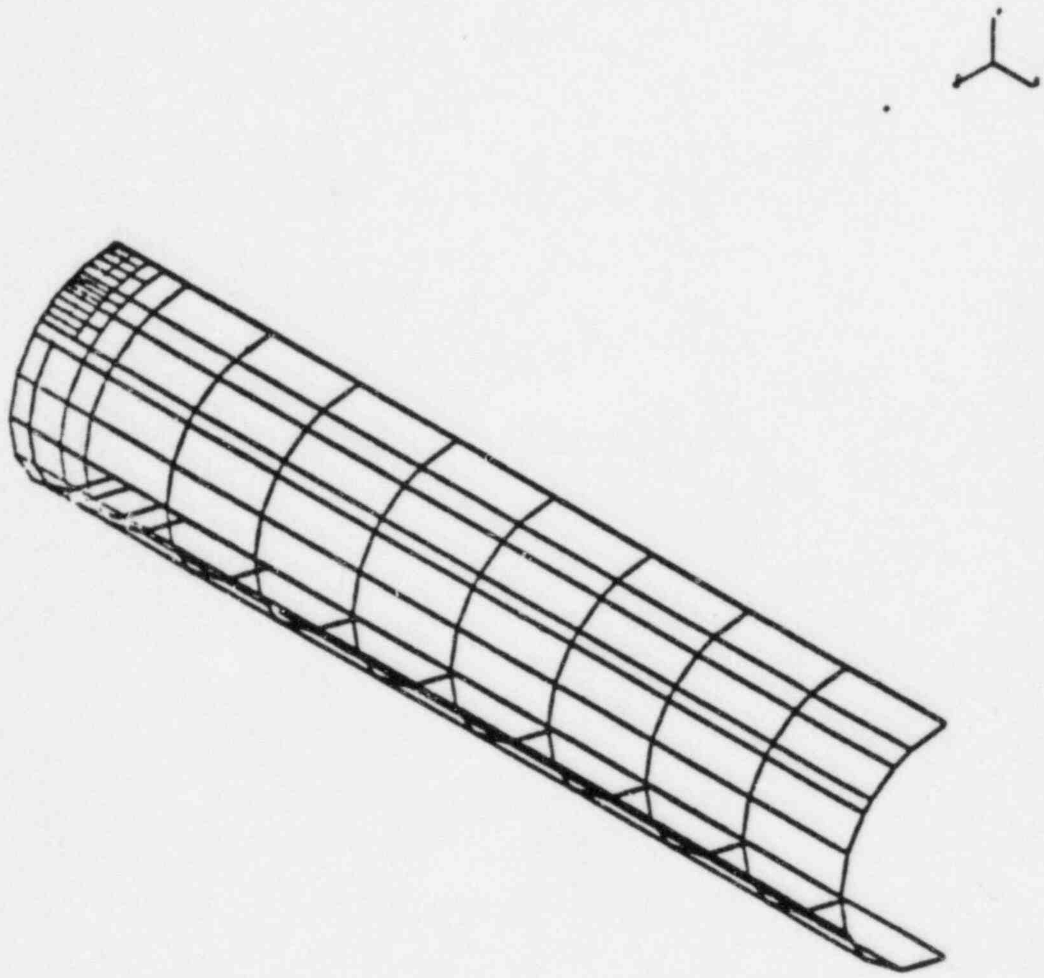


Figure 5-2   Finite Element Model

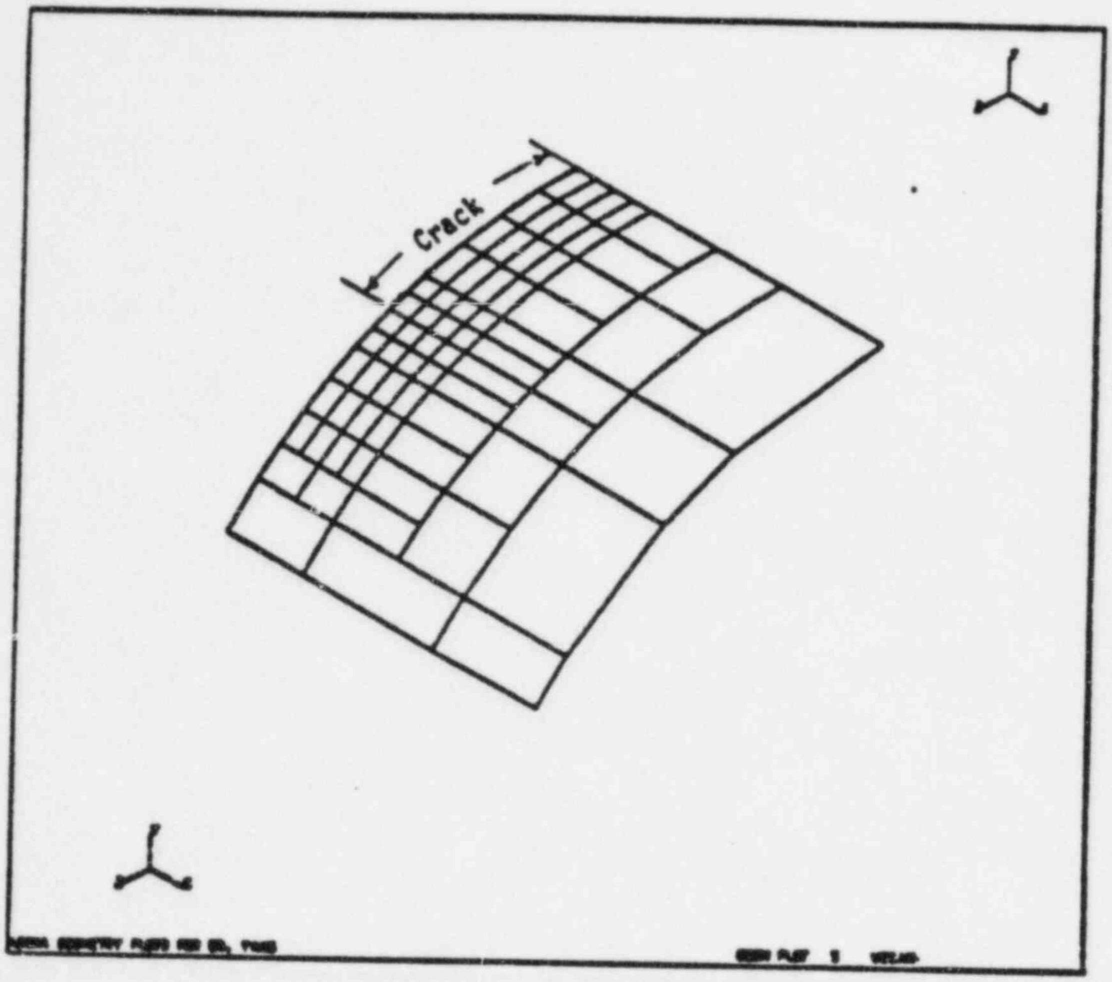


Figure 5-3 Details of Finite Element Model in the Vicinity of Crack

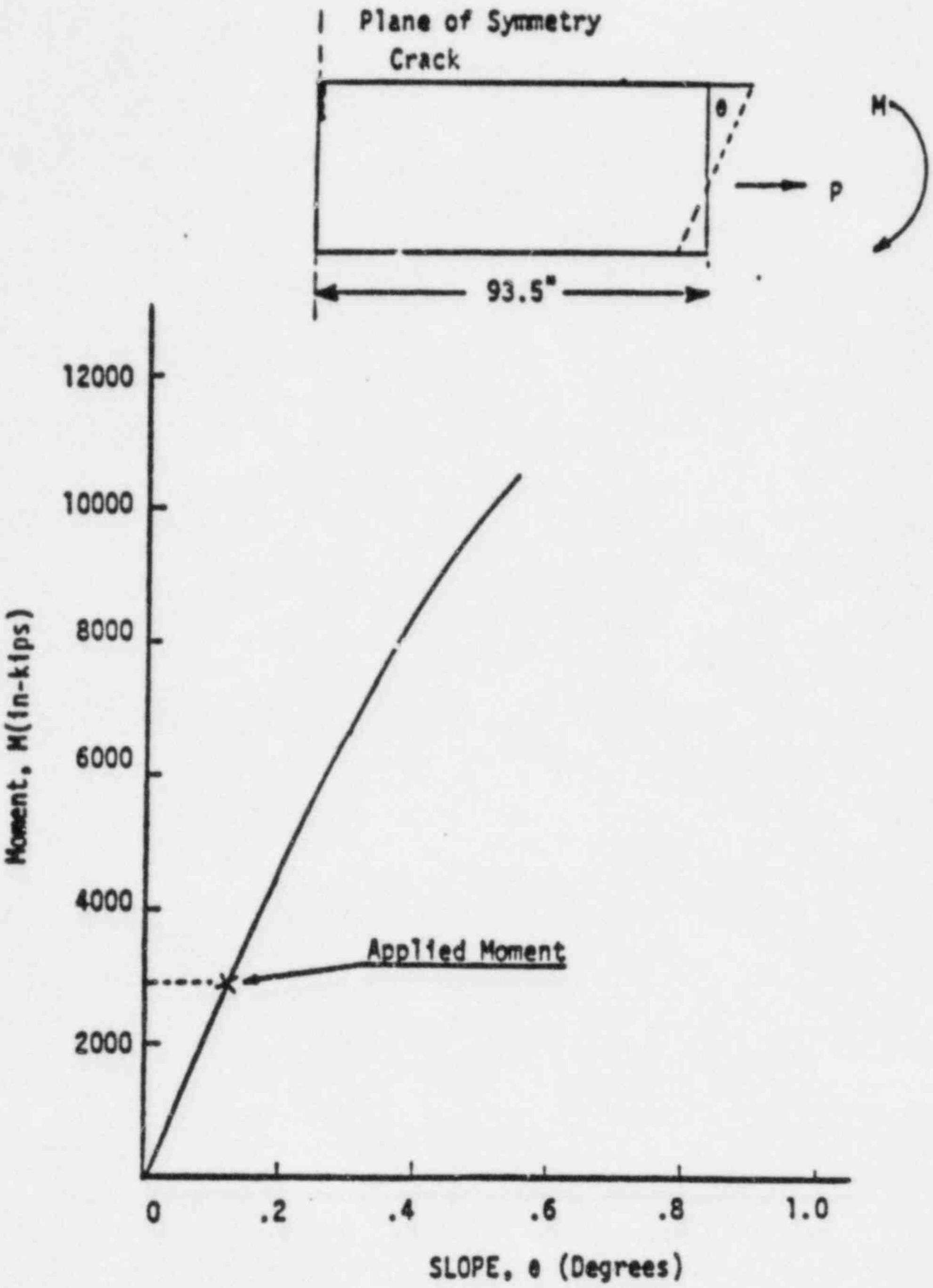


Figure 5-4 Variation of Pipe End Slope with Applied Static Moment

6. FRACTURE MECHANICS EVALUATION OF MAIN STEAM LINE PIPE

6.1 Global Criteria

The general global criterion specified in section 4-2 requires that

$$W_a < W_p \quad (4\text{-}1)$$

where W_a = applied generalized load

W_p = calculated generalized plastic load.

For the static loading, ADINA finite element elastic plastic calculations show that the moment carrying capacity of the main steam line pipe containing a 10 inch through wall circumferential flaw is at least 10580 in-kips for a given steam pressure of 710 psi.

The generalized plastic load W_p is then at least 10580 in-kips for the given pressure of 710 psi. The applied generalized load W_a is 2880 in-kips for a given pressure of 710 psi. Therefore, the criterion given by equation 4-1 is satisfied.

6.2 Local Criteria

The general local criteria specified in Section 4-3 require that

$$J < J_{IN} \quad (4\text{-}2)$$

An estimate for the J integral can be obtained based on the following approximate relation, using the finite element analysis results:

$$J = \frac{\pi}{8r} [E^1 \delta_e^2 + E^T \delta_p^2] \quad (6\text{-}1)$$

where δ_e, δ_p = elastic and plastic displacement of the quarter-point node behind the crack tip, respectively.

r = distance between crack tip and the quarter-point node

E^1 = E, $E/1-\nu^2$ for plane stress and plane strain, respectively.

E^T = Strain-hardening modulus.

It should be noted that the above expression is strictly applicable only to a crack in a two-dimensional bilinear elastic medium. One would expect that this would give reasonable estimate for the bilinear elastic-plastic case where unloading does not occur.

For the present case, the J-integral value of 51 in-lb/in^2 is obtained corresponding to a load of 3034 in-kips using equation (6-1). Since the maximum applied generalized load [5-2] is 2280 in-kips (<3034) this calculated J integral value of 51 in-lb/in^2 can be considered as the maximum resulting J from the applied loads.

The value of J_{IN} for the SA106B is available at 425°F and varies between 674 in-lb/in^2 (6-1) and 1096 in-lb/in^2 (unpublished Westinghouse results). Clearly, the calculated J value of 51 in-lb/in^2 is very small compared to J_{IN}. Therefore, equation (4-2) is satisfied.

7. SUMMARY AND CONCLUSIONS

7.1 Summary

The objective of this investigation was to examine mechanistically, under realistic and yet sufficiently conservative assumptions, whether a crack which was assumed to appear instantaneously, in the main steam line pipe of San Onofre Unit 1, would become unstable and lead to a full circumferential break when subjected to the worst possible combination of plant loadings. The scope of this investigation included:

- Postulating a circumferential through-wall flaw.
- Performing static elastic-plastic finite element analysis of the cracked pipe using the ADINA Code.
- Evaluation of global criteria based on plastic instability load.
- Obtain an estimate of J integral to evaluate the local criterion.

7.2 Conclusions

Based on the analysis the following conclusions are drawn.

- A 10 inch long through wall circumferential flaw in the main steam line pipe will be stable globally and locally.

- Under the worst combination of loadings including the effects of design basis earthquake, thermal expansion, dead weight and turbine valve closure, a realistically postulated flaw will not propagate around the circumference and cause a guillotine break.

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2-2 Fatigue Crack Growth Evaluation for San Onofre Unit 1 Main Steam Line Pipe, WCAP 9808, October 1980.

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3-2 1977 Annual Book of ASTM Standards, Part 1, Steel Piping, Tubing and Fittings.

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4-2 Mechanical Fracture Predictions for Sensitized Stainless Steel Piping with Circumferential Crack, EPRI-NP-192, September 1976.

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WCAP-9808

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FATIGUE CRACK GROWTH EVALUATION FOR SAN ONOFRE UNIT 1 MAIN STEAM LINE PIPE

By

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October, 1980

PREPARED BY WESTINGHOUSE FOR SOUTHERN CALIFORNIA EDISON COMPANY

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Work Performed Under SCFN-2020

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PREFACE

This report has been technically reviewed and the calculations checked.

W. T. Kaiser
W. T. Kaiser

ABSTRACT

A fatigue crack growth evaluation for the San Onofre Unit 1 main steam line pipe is presented in this report. The analysis is based on postulated initial flaws of 0.125, 0.250 and 0.375 inch depths. The postulated flaws are oriented in the circumferential direction. The fatigue crack growth analysis has been conducted in the same manner as suggested by Section XI, Appendix A of the ASME Boiler and Pressure Vessel Code.

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SECTION 1

INTRODUCTION

The linear elastic fracture mechanics approach to the design against failure is basically a stress intensity factor consideration in which criteria are established for fracture instability in the presence of a crack-like flaw. Consequently, a basic assumption in employing the fracture mechanics technology is that a crack or crack-like defect (either due to postulation or defective manufacture) exists in the structure being evaluated. By using the stress intensity factor, K_I, all pertinent variables (flaw size, structural geometry and nominal stress) can be condensed into one parameter.

A necessary ingredient in the concept of fracture mechanics is knowledge of the present crack size. A fatigue crack growth evaluation will determine the growth of a flaw (either postulated or discovered during an in-service inspection) through end of life from the initial state. This growth is a result of variations in the crack tip stress field due to coolant pressure and temperature changes during transients. The procedure for such a fatigue analysis involves finding the crack growth during each transient and adding this growth to the initial crack size.

This report presents the fatigue crack growth evaluation of the main steam line for San Onofre Unit 1, and the analysis results will be used in the mechanistic pipe break study of the main steam line with a postulated circumferential flaw. Since a circumferential flaw represents the most severe flaw orientation for the pipe break analysis, only circumferential flaws are considered in this fatigue crack growth evaluation.

The fatigue crack growth analyses presented herein were conducted in the same manner as suggested by Section XI, Appendix A of the ASME Boiler and Pressure Vessel Code[1]. The analysis procedure involves assuming an initial flaw exists at some point and predicting the growth of that flaw due to an imposed series of stress transients. The growth of a crack per loading cycle is dependent on the range of applied stress intensity factor ΔK_I, by the following relation:

$$\frac{da}{dN} = C_0(\Delta K_I)^n \qquad (1\text{-}1)$$

where "C_0" and the exponent "n" are functions of material properties.

The input required for a fatigue crack growth analysis is basically the information necessary to calculate the parameter ΔK_I, which depends on crack and structure geometry and the range of applied stresses in the area where the crack exists. Once ΔK_I is calculated, the growth due to that particular cycle can be calculated by equation (1-1). This increment of growth is then added to the original crack size, and the analysis proceeds to the next transient. The procedure is continued in this manner until all the transients known to occur in the period of evaluation have been analyzed.

Crack tip stress intensity factors are calculated using semi-elliptic surface flaw expressions. Mechanical stresses and the thermal stress distribution through the thickness of the pipe are used in the calculation of the stress intensity factor K_I.

The technique presented in this report is to determine the final crack depth for an assumed initial flaw postulated in the pipe wall. The postulated initial crack depths range from 0.125 in. to 0.375 in. and are considered to realistically encompass the range of flaws that could be present.

SECTION 2

THERMAL AND STRESS ANALYSIS

This section presents the results of the transient thermal and stress analysis of the main steam line for San Onofre Unit 1. The purpose of this analysis is to determine the stresses in the pipe due to the transient thermal and mechanical loads identified in the applicable pressure vessel equipment specification.

2-1 ASSUMPTIONS

The geometry of the main steam line is shown in figure 2-1. The interior surface of the pipe is in contact with the steam, and the resulting heat transfer coefficient is estimated as 400 Btu/hr-ft^2-°F based on the Dittus-Boelter[2] forced convection correlation for all of the reactor design transients. The outside surface is assumed to be insulated.

2-2 THERMAL ANALYSIS - TEMPERATURES AND STRESSES

The heat transfer analysis for each of the transients was carried out by an explicit finite difference heat transfer analysis[3]. The temperature profiles generated by this analysis were then used to calculate thermal stresses. The equations for thermal stress in a hollow cylinder from Timoshenko and Goodier[4] were used:

$$\text{radial stress} = \sigma_r = \left(\frac{\alpha E}{1-\nu}\right)\frac{1}{r^2}\left(\frac{r^2 - a^2}{b^2 - a^2}\int_a^b Trdr - \int_a^r Trdr\right) \qquad (2\text{-}1)$$

$$\text{tangential stress} = \sigma_\theta = \left(\frac{\alpha E}{1-\nu}\right)\frac{1}{r^2}\left(\frac{r^2 + a^2}{b^2 - a^2}\int_a^b Trdr + \int_a^r Trdr - Tr^2\right) \qquad (2\text{-}2)$$

$$\text{axial stress} = \sigma_z = \left(\frac{\alpha E}{1-\nu}\right)\left(\frac{2}{b^2 - a^2}\int_a^b Trdr - T\right) \qquad (2\text{-}3)$$

where r = radial position

T = temperature as function of r; $T \rightarrow T(r)$

a = inner radius of the pipe

b = outer radius of the pipe

ν = Poisson ratio

αE = the product of the coefficient of thermal expansion and the modulus of Elasticity

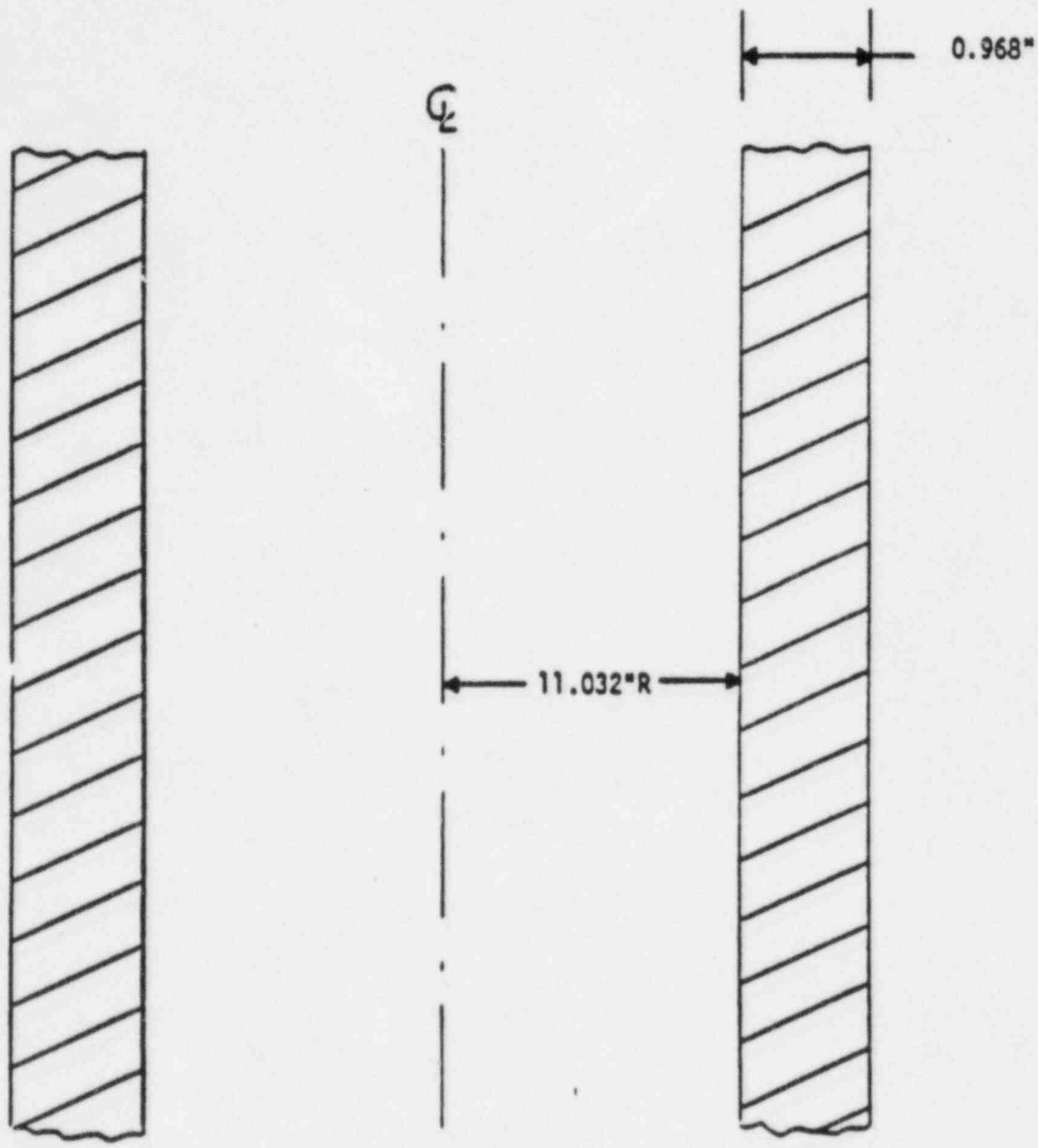

Figure 2-1 San Onofre Unit 1 Main Steam Line

The integrals in equations (2-1) through (2-3) are evaluated numerically to provide the necessary thermal stresses for each of the transients analyzed.

2-3 MECHANICAL STRESS ANALYSIS

The mechanical loading for the pipe results only from internal pressure, and since this is not a discontinuity region, the resulting stresses in the steam line were calculated in closed form:

$$\sigma_{h_i} = P \frac{(b^2 + a^2)}{(b^2 - a^2)} \tag{2-4}$$

$$\sigma_{h_o} = P \frac{2a^2}{(b^2 - a^2)} \tag{2-5}$$

$$\sigma_{a_i} = \sigma_{a_o} = P \frac{a^2}{(b^2 - a^2)} \tag{2-6}$$

where

P = internal pressure

a = inner wall radius

b = outer wall radius

σ_h = hoop stress

σ_a = axial stress

i = inside surface

o = outside surface

The thermal and mechanical stresses are combined, and then linearized through the steam line wall thickness to allow for calculation of the applied stress intensity factor at any given time in a transient, as will be described in detail in Section 4.

In San Onofre Unit 1 plant, the material used in the main steam line is SA106 Grade B. Table 2-1 lists the mechanical and physical properties from ASME Section III[5] used in the analysis.

TABLE 2-1

MATERIAL PROPERTIES

Property (500°F)	Material SA106 Grade B
Young's Modulus (psi)	26.4×10^6
Density (lb/in.^{3})	0.281
Conductivity (Btu/hr-in.-°F)	2.217
Heat Capacity (Btu/lb-°F)	0.132
Coefficient of Thermal Expansion (in/in.-°F)	8.18×10^{-6}
Poisson's Ratio	0.30

SECTION 3
DESIGN TRANSIENTS

The design transients used in the fatigue evaluation of the main steam line pipe are given in table 3-1. The transient conditions selected for this evaluation are based on conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. These are representative of transient conditions which are considered to occur during plant operation and are sufficiently severe or frequent to be of significance to component cyclic behavior. Further, these are regarded as a conservative representation of transients which, when used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant.

The total number of cycles for each operating transient exclusive of the pre-operational test cycles has been assumed to be evenly divided over the 40-year operating life of the plant. The assumed schedular distribution of the reactor operating transients is shown in table 3-2.

TABLE 3-1

SECONDARY SIDE DESIGN TRANSIENTS

TRANSIENT	NUMBER OF OCCURRENCES
Hot Standby	18300
Plant Loading and Unloading at 5% of Full Power/Minute	18300
Small Step Load Increase & Decrease	2000
Large Step Load Decrease	200
Loss of Power	40
Partial Loss of Flow	80
Loss of Load	80
Reactor Trip	400
TEST CONDITIONS	
Secondary Side Cold Hydro	5

TABLE 3-2

SCHEDULE OF SECONDARY SIDE OPERATING TRANSIENTS[a]

TRANSIENT	NUMBER OF OCCURRENCES			
	5 Events Per Year	2 Events Per Year	1 Event Per Year	1 Every 4th Year
Hot Standby	91	1	-	2
Plant Loading and Unloading at 5% of Full Power/Minute	91	1	-	2
Small Step Load Increase & Decrease	10	-	-	-
Large Step Load Decrease	1	-	-	-
Loss of Power	-	-	1	-
Partial Loss of Flow	-	1	-	-
Loss of Load	-	1	-	-
Reactor Trip	2	-	-	-

[a] This table does not include preoperational test cycles since they occur prior to plant operation.

SECTION 4

STRESS INTENSITY FACTOR CALCULATIONS

This section describes the method of calculating the stress intensity factor K_I using membrane and bending stresses. The stresses are determined by stress analysis as described in Section 3. Stresses resulting from pressure and thermal transients are considered in calculating the stress intensity factors. The actual stress distribution through tho pipe wall is conservatively approximated by using the linearization technique illustrated in figure 4-1.

In this analysis, a circumferential flaw on the inside surface of the pipe is postulated. Crack depths varying from 0.125 inch to 0.375 inch have been included to determine the sensitivity of the results to the initial assumed flaw depth. The initial flaw depth of 0.375 inch represents a 39% through wall flaw. A semielliptical configuration with length-to-depth ratio of six and its major axis on the surface is assumed for the shape of the flaw as shown by figure 4-2.

4-1 K_I EXPRESSION

The stress intensity factor K_I at the point of maximum depth is calculated from the membrane and bending stresses using the following equation from Section XI of the ASME Code[1]:

$$K_I = \sqrt{\frac{\pi a}{Q}}\ (\sigma_m M_m + \sigma_b M_b) \qquad (4\text{-}1)$$

where

$\sigma_{m,b}$ = membrane and bending stress, respectively

a = minor semiaxis (flaw depth)

Q = flaw shape parameter including a plastic zone correction factor for plane strain conditions, (see figure 4-3).

$$Q = [\phi_1^2 - 0.212\ (\sigma/\sigma_{ys})^2]$$

$$\phi_1 = \int_0^{\pi/2} \sqrt{1 - \left(\frac{b^2 - a^2}{b^2}\right) \sin^2\phi}\ d\phi$$

σ_{ys} = yield strength of material

σ = $\sigma_m + \sigma_b$

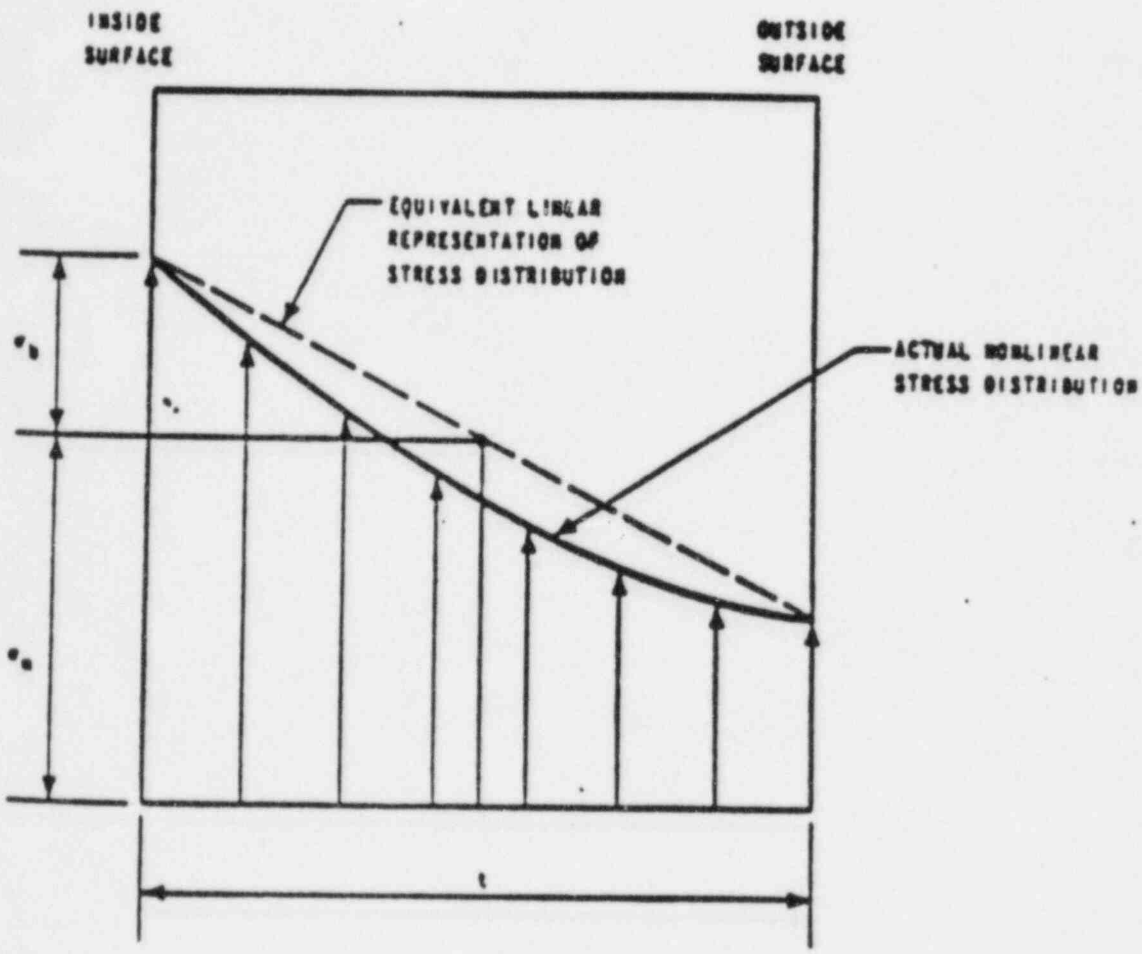

Figure 4-1. Linearized Representation of Stresses

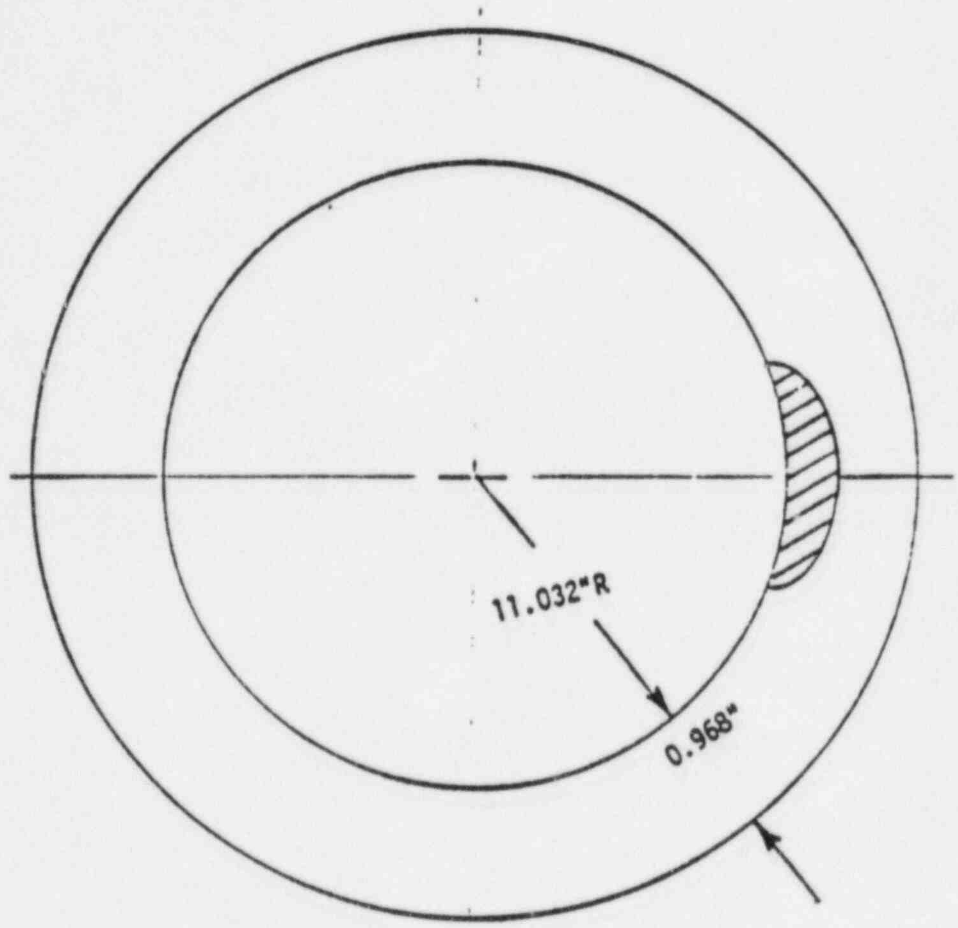

Figure 4-2. Postulated Flaw - Circumferential Semielliptical Surface Flaw

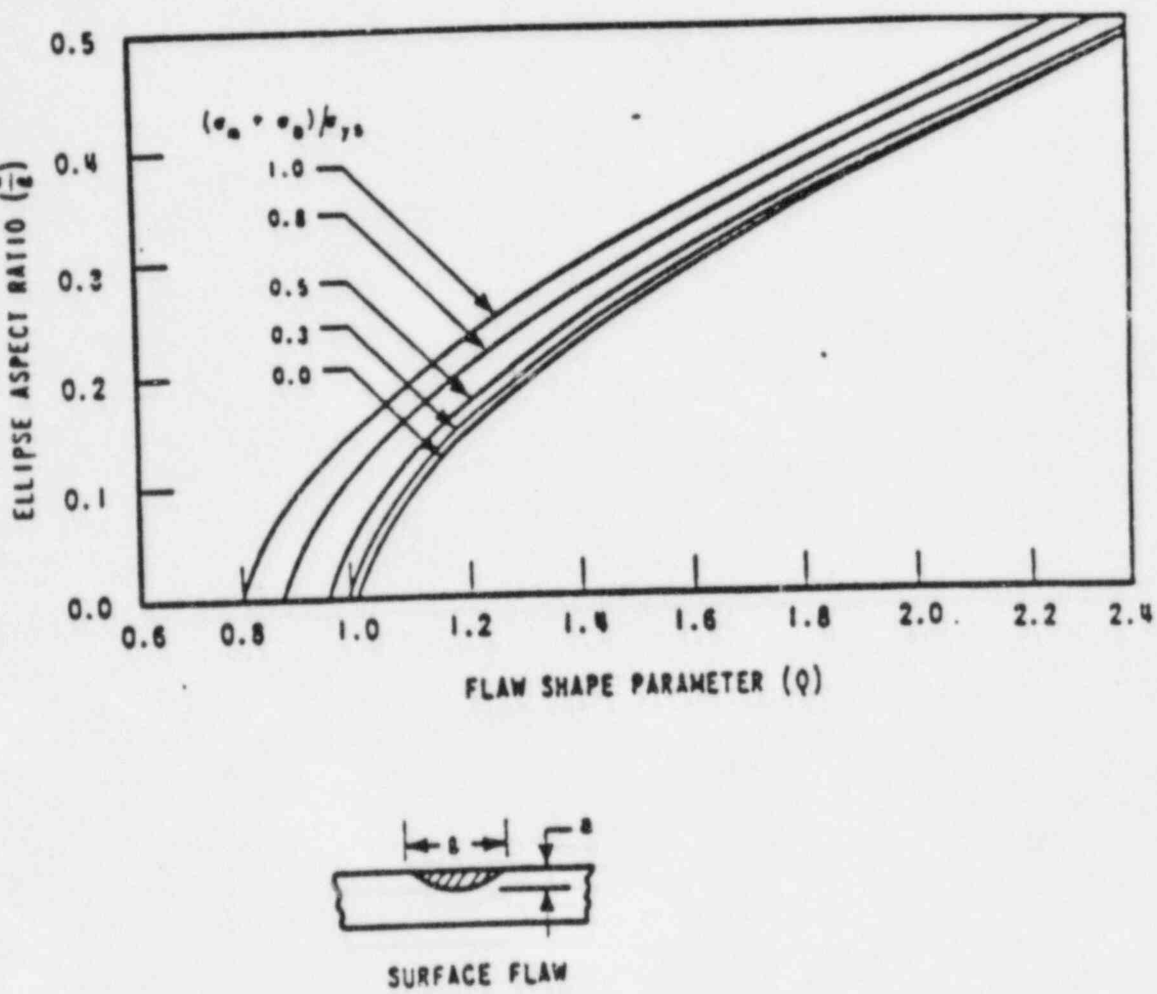

Figure 4-3 Shape Factors for Surface Flaws

b = major semiaxis (flaw length/2)

ϕ = parametric angle of the ellipse

M_m = correction factor for membrane stresses (see figure 4-4)

M_b = correction factor for bending stresses (see figure 4-5)

The inside and outside stresses for each transient for the three regions analyzed are given in table 4-1. The stress values which yield both the maximum and the minimum K_I values for each transient are listed.

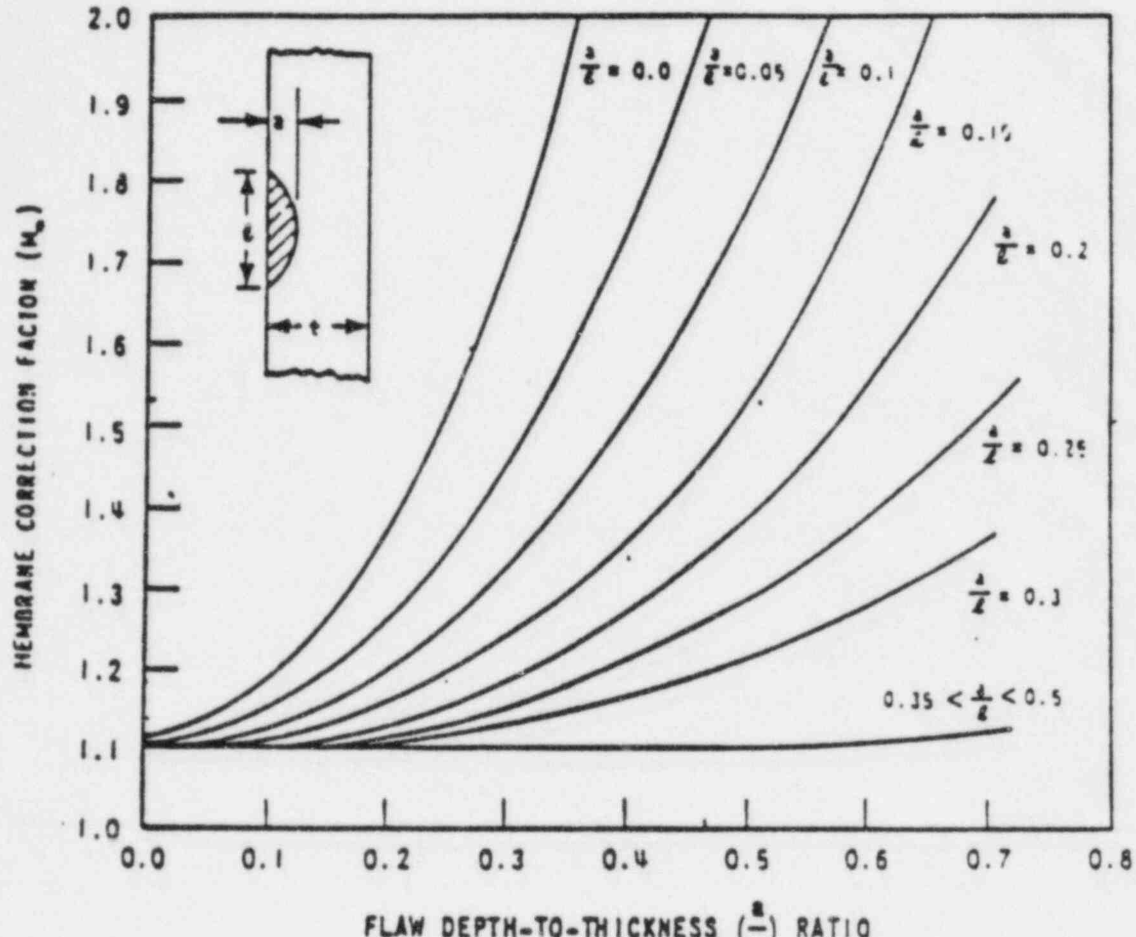

Figure 4-4 Membrane Correction Factors for Surface Flaws

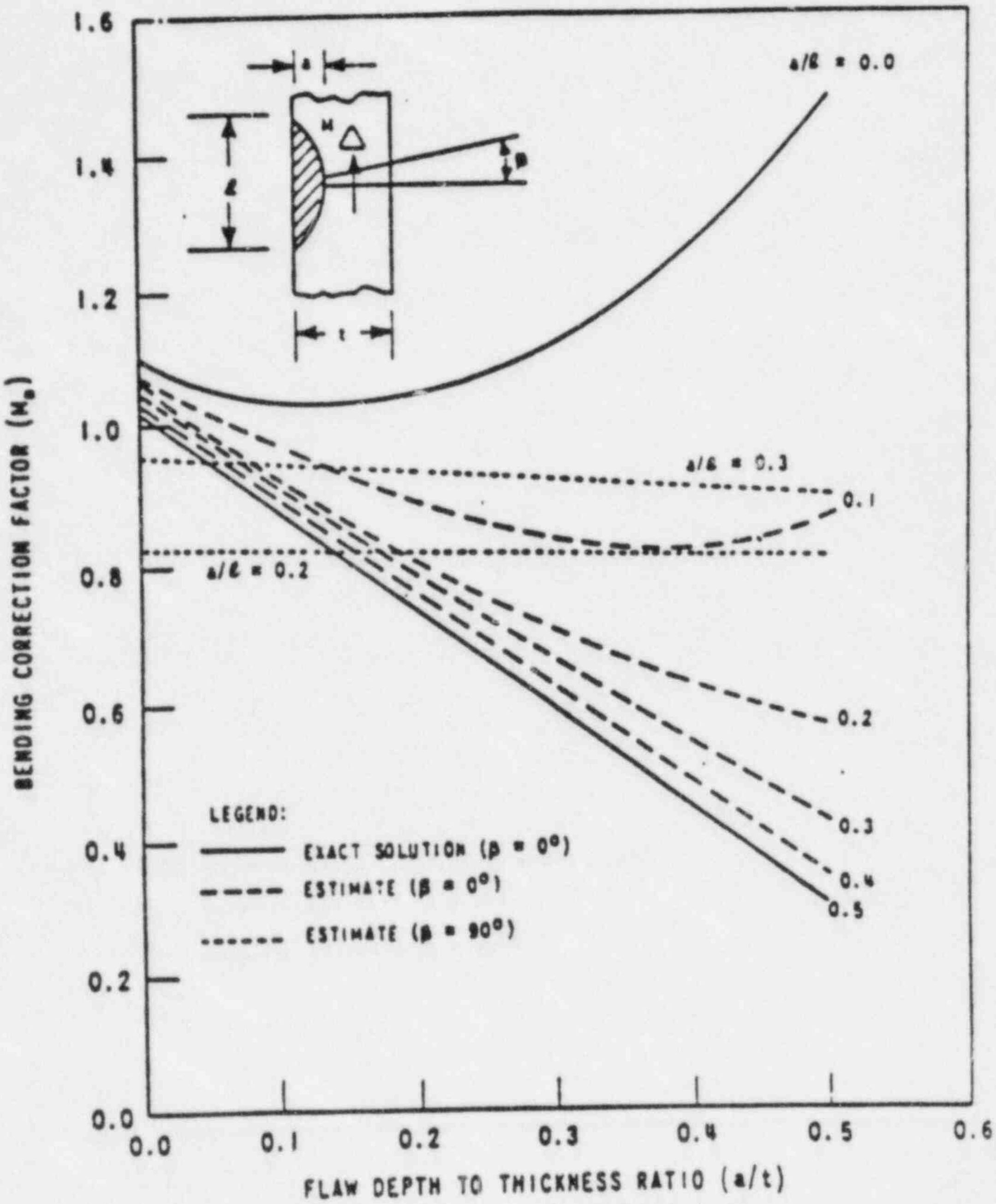

Figure 4-5 Bending Correction Factor for Surface Flaws

TABLE 4-1 INSIDE AND OUTSIDE LONGITUDINAL STRESSES

Transient	σ Inside[a] (ksi)	σ Outside[a] (ksi)	σ Inside[b] (ksi)	σ Outside[b] (ksi)
Hot Standby	5.164	5.100	3.933	3.939
Unit Loading and Unloading	5.096	5.013	3.421	3.715
Small Step Load Increase & Decrease	4.024	3.866	3.638	3.785
Large Step Load Decrease	4.340	4.248	3.452	4.541
Loss of Power	5.184	4.612	3.817	3.817
Partial Loss of Flow	4.511	4.511	3.341	4.798
Loss of Load	4.996	4.996	2.773	6.054
Reactor Trip	4.038	3.106	1.705	1.705
Cold Hydro	8.092	8.092	0.000	0.000

[a] These stress values lead to maximum K_I

[b] These stress values lead to minimum K_I

SECTION 5

FATIGUE CRACK GROWTH RATE

The growth of a crack per loading cycle is dependent on the range of applied stress intensity factor ΔK_I, by the following relations:

$$\frac{da}{dN} = C_o \, (\Delta K_I)^n \qquad (5\text{-}1)$$

where "C_o" and the exponent "n" are functions of material properties.

5-1 ASME SECTION XI CRACK GROWTH LAW FOR WATER REACTOR ENVIRONMENT

The upper bound curve from ASME Section XI[1] for fatigue crack growth analysis shown by figure 5-1 is considered applicable to SA106 Grade B material in this analysis based on the following justification provided in Reference [6]. The reference law in the ASME Code Section XI was designed to be applicable to carbon and low alloy steels to minimum yield strengths less than or equal to 50 ksi, although no data were available at the time of its inception to support such a wide application [6]. Data are now available to demonstrate that medium strength carbon and low alloy steels do indeed have very similar behavior in a water environment. Besides the original test materials of A508 Cl 2 and A533B Cl 1 there are test results available for ASTM A516 GR-70 steel in light water reactor environment which agree well with the reactor vessel steels[7]. This steel has a minimum specified yield strength of 38 ksi. Further data in water environments have been obtained by Soctt [8] on lower strength steels and Vosikovsky [9] on higher strength line pipe steel (65 ksi minimum yield strength). This information suggests that the reference curve should have applicability to all carbon and low alloy steels with minimum yield strength less than 65 ksi.

The reference law in the ASME Code Section XI is represented by the following expression.

$$\frac{da}{dN} = (0.3795 \times 10^{-3}) \, \Delta K_I^{\,3.726} \qquad (5\text{-}2)$$

where, $\frac{da}{dN}$ = Crack growth rate, micro-inches/cycle

ΔK_I = stress intensity factor range, ksi$\sqrt{\text{in}}$ = $(K_{I\,max} - K_{I\,min})$

$K_{I\,max}$, $K_{I\,min}$ = Maximum and Minimum K_I respectively computed during the transient

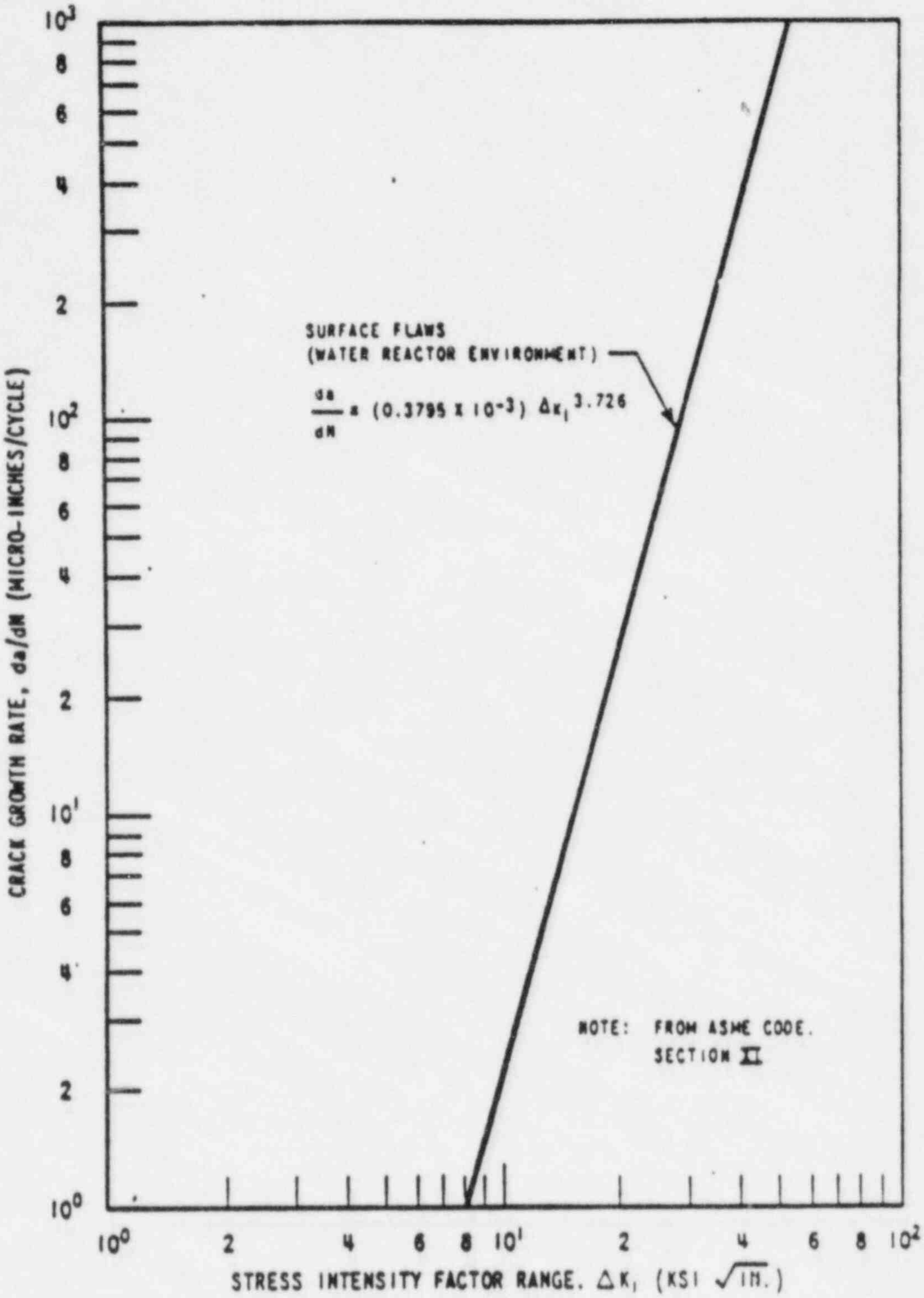

Figure 5-1. Fatigue Crack Growth Reference Law for Reactor Vessel Steels

SECTION 6

FATIGUE EVALUATION

The fatigue crack growth analysis presented herein has been conducted in the same manner as suggested by Section XI Appendix A of the ASME Boiler and Pressure Vessel Code[1]. The analysis procedure involves postulating an initial circumferential flaw and predicting the growth of that flaw due to an imposed series of stress transients. The input required for a fatigue crack growth analysis is basically the information necessary to calculate the parameter ΔK_I which depends on crack and structural geometry and the range of applied stresses in the area where the crack exists. Once ΔK_I is calculated, the growth due to that particular cycle can be calculated by equation 4-1. This increment of growth is then added to the original crack size, and the analysis proceeds to the next transient. The procedure is continued in this manner until all the transients known to occur in the period of evaluation have been analyzed.

In order to determine the maximum potential for fatigue crack growth of the postulated flaw in the pipe during normal operation, a cumulative fatigue crack growth analysis is performed. All design transients are considered in chronological order according to the assumed schedule prescribed in table 3-2. Stress intensity factors are determined for each transient using the bounding semielliptical flaw model and the method for K_I determination outlined in Section 4. Each transient is evaluated in the following manner:

1) Determine the maximum range of K_I fluctuation (ΔK_I associated with the transient).
2) Find the incremental flaw growth (Δa) corresponding to ΔK_I from the fatigue crack growth rate data.
3) Update the flaw size by assuming the flaw grows to a geometrically similar, larger flaw with a minor half axis ($a + \Delta a$).
4) Proceed to the next transient.

The above procedure, after all transients have been considered, yields the expected end-of-life flaw size (a_f). The procedure has been automated and the crack growth results are obtained for 0.125 inch, 0.250 inch, and 0.375 inch postulated initial flaw depths. The stress intensity factor ranges (ΔK_I) associated with the transients are presented in table 6-1.

TABLE 6-1

STRESS INTENSITY FACTOR RANGES (ΔK_I)

No.	Transient	Postulated Flaw Depth (in.)	ΔK_I (ksi$\sqrt{in}$)
1	Hot Standby	.125	.765
		.250	1.143
		.375	1.546
2	Unit Loading and Unloading	.125	1.024
		.250	1.509
		.375	2.016
3	Small Step Load Increase and Decrease	.125	0.223
		.250	0.312
		.375	0.397
4	Large Step Load Decrease	.125	.485
		.250	.639
		.375	.765
5	Loss of Power	.125	.820
		.250	1.189
		.375	1.564
6	Partial Loss of Flow	.125	.644
		.250	.859
		.375	1.039
7	Loss of Load	.125	1.194
		.250	1.543
		.375	1.816

TABLE 6-1 (cont'd)

STRESS INTENSITY FACTOR RANGES (ΔK_I)

No.	Transient	Postulated Flaw Depth (in.)	ΔK_I (ksi$\sqrt{in}$)
8	Reactor Trip	.125	1.402
		.250	2.035
		.375	2.681
9	Cold Hydro	.125	5.068
		.250	7.606
		.375	10.326

SECTION 7
RESULTS AND CONCLUSIONS

A fatigue crack growth analysis has been carried out for the main steam line pipe of San Onofre Unit 1, and the results of the analys's are summarized in table 7-1. This table presents the fatigue crack growt', results for a range of postulated flaw depths oriented :ircumferentially. The postulated flaws are assumed to be six times as lc j as they are deep. Based on these results, it is concluded that growt' by fatigue is negligible.

TABLE 7-1

RESULTS OF FATIGUE CRACK GROWTH EVALUATION

Postulated Initial Crack Depth (in)	Crack Depth (in) After Year			
	10	20	30	40
0.125	0.12500	0.12501	0.12501	0.12501
0.250	0.25002	0.25003	0.25004	0.25005
0.375	0.37505	0.37508	0.37511	0.37515

SECTION 8

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