June 13, 1997

MEMO TO: PD IV-1 File

FROM: Tom Alexion ORIG SIGNED BY:
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

SUBJECT: LICENSEE'S 10 CFR 50.59 EVALUATION OF MAIN STEAM LINE BREAK (MSLB) REANALYSIS AND EFFECT ON ISOLATION VALVE CUBICLE (IVC) AND REACTOR CONTAINMENT BUILDING (RCB) (TAC NOS. M98914 AND M98915)

I requested the licensee to provide the above subject document. The purpose of this memo is to place this information in the public document room.

Docket Nos. 50-498 and 50-499

Attachment: As stated

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Document Name: STP98914.DOC

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 13, 1997

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FROM: TWA Tom Alexion
Project Directorate IV-1
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Docket Nos. 50-498 and 50-499

Attachment: As stated

PORC Review Cover Sheet
0PAP01-ZA-0104-1
(Page 1 of 1)
(Sample)

Originating Document No. USQE 95-0013 Revision No. 0

TITLE CN-1962 Revise analysis of the Main Steam Line Break in the IVC and RCB

The PORC has reviewed this item and has determined that (check as appropriate):

It ___ does ✓ does NOT involve an UNREVIEWED SAFETY QUESTION.

It ___ does ✓ does NOT adversely impact plant nuclear safety.

It ___ does ✓ does NOT adversely impact the health and safety of plant personnel or the public.

It ✓ does ___ does NOT require further review by the Plant Mgr, the NSRB, or other individuals/groups.

✓ Plant Manager Unit 1 2 ✓ NSRB ___ other (specify below)

REMARKS

ST1-95-004328-01

The PORC recommends this item for:

✓ APPROVAL ___ DISAPPROVAL ___ OTHER ___ PORC MEETING NO. 95-017

Completed by [signature] DATE 3/31/95
PORC Secretary

This form, when completed, SHALL be retained in accordance with the retention requirements of the originating document.

☐ UNIT #1 ☐ PROCEDURE ☐ PLANT MODIFICATION ☐ ECNP ☐ DCN
☐ UNIT #2
☐ BOTH ☐ UFSAR ON ☐ OTHER ______

ORIGINATING DOCUMENT NO. Licenisng Change Notice 1962 REV. NO. 0

SUMMARY DESCRIPTION OF CHANGE: UFSAR changes revising the analysis of the main steam line break in the IVC and RCB

REASON FOR CHANGE: Revised analysis to support JCO 93-0004, "MSLB Blowdown Model for the IVC

PRELIMINARY SCREENING

	YES	NO
1. Does the proposed change represent a change to the Plant Technical Specifications?	☐	☒

2. If an Unreviewed Safety Question known is known to be associated with the subject change, then further screening is not required; refer to IP-1.19Q.

If "Yes refer to IP-1.19Q. Further screening is not required.

Does the proposed change represent:	YES	NO
3. A change to correct a typographical, editorial or drafting error?	☐	☒
4. A change which is identical to and addressed in its entirety by an existing approved 10CFR50.59 Screening/USQE?	☐	☒
5. A procedure change in which the format or text changed without changing actions or intent?	☐	☒
6. A spare or replacement part/component change with an equivalent part/component? (See Section 2.16 for a definition of equivalent)	☐	☒

If all answers to the above questions are "NO" perform the final screening and mark N/A in the approval blocks below.
If the answer to any question (3) through (6) is "YES" a final screening is not necessary.
Sign approval blocks below and discard pages 2 through 4.
Provide an explanation/justification and references if any of items (3) through (6) is answered "YES".

Prepared by: N/A ______ Originator ______ Date

Approved by: N/A ______ Section Supervisor ______ Date

ORIGINATING DOCUMENT LCN 1962

FINAL SCREENING

In response to the questions below, if the change involves something that is not described in the SAR and is not part of the licensing basis as shown by a review of NRC-published documents, then "NO" is appropriate. However, this decision must be clearly documented with adequate technical justification.

The phrase "not part of licensing basis" implies that the subject matter was not used by NRC to issue or maintain the operating license or amendments; and is determined by examination of the Licensing Docket and the following (as applicable):

Safety Analysis Report	Training and Qualification Program
Environmental Report	Final Environmental Statement
Fire Hazards Analysis Report	Safety Evaluation Report
Physical Security Plan	Standard Review Plan
Safeguards Contingency Plan	Correspondence
Operations Quality Assurance Plan	Emergency Plan
Previously Approved USQ Evaluations	

		YES	NO
1.	Does the subject of this review involve a change to the facility as described in the Safety Analysis Report?	☒	☐
2.	Does the subject of this review involve a change to the procedures as described in the Safety Analysis Report?	☐	☒
3.	Does the subject of this review propose the conduct of tests or experiments not described in the Safety Analysis Report?	☐	☒
4.	Does the proposed change affect conditions or bases assumed in the Safety Analysis Report or safety-related functions of equipment/systems, even though the proposed change does not entail any physical change in existing structures, systems, or procedures as described in the SAR?	☒	☐

If any answer is affirmative, complete the screening form and perform an Unreviewed Safety Question Evaluation.

If all answers are negative, no Unreviewed Safety Question Evaluation is required.

All questions require adequate technical justification.

ORIGINATING DOCUMENT: LCN 1962

TECHNICAL JUSTIFICATION SHOULD INCLUDE THE SAFETY IMPLICATIONS OF THE CHANGE AND OTHER INFORMATION SUPPORTING ALL ANSWERS.

BRIEFING DESCRIPTION AND TECHNICAL JUSTIFICATION OF THE CHANGE: The proposed change is a revision to the UFSAR mass and energy release steam line break analysis for the IVC and RCB. JCO93-0004 Revision 5 contains additional information concerning this change.

(use additional pages as necessary)

Interdiscipline Coordination Required? ☒ Yes ☐ No
If "yes, obtain appropriate concurrence.

☐ Risk and Reliability Analysis ☐ Thermal Hydraulics ______ ☐ Reactor Engr. ______

☒ Civil RA 3/31/95 ☐ Mech ______ ☐ Elect ______ ☐ I&C ______ ☐ EQ ______ ☐ Other ______

Prepare by: Charles R Albury 3/30/95
Originator Date

Approved by: D.C. Logan 3/31/95
Department Manager Date

ORIGINATING DOCUMENT LCN 1962

The following documents/attributes have been reviewed as part of the 10CFR50.59 final screening process.

Documents	Sections Reviewed
UFSAR	3.6.A, 6.2.1.3
Technical Specifications	
Safety Evaluation Report (SER)	6.2.1.2
Fire Protection (FHAR)	
Environmental Report (ER)	
Security Plan	
Emergency Plan	
Offsite Dose Calculation Manual (ODOM)	
Final Environmental Statement	
Core Operating Limits Report (COLR)	
Operations Quality Assurance Plan	
Other See JCO 93-004, Rev 5	

Attributes	Check if Reviewed
Environmental Qualification	X
Seismic Design	
Personnel Radiation Exposure	
Missile Protection	
Containment Integrity	
Single Failure Criteria	
Electrical Separation (RG 1.75)	
Heavy Loads	
High Energy Line Break Accident Analysis	X
Control Room Habitability	
Internal Flooding	
Plant Chemistry	
Human Factors	
Probabilistic Safety Assessment	
Other	

NOTE: If Attributes are identified in the originating document this section need not be completed.

Unreviewed Safety Question Evaluation # 95-0013 Revision No. 0

ORIGINATING DOCUMENT: LCN 1962 REV. NO. 0

NOTE: Attach 10CFR50.59 Screening Form or License Compliance Form to this USQE.

NOTE: Use additional sheets as necessary to provide the bases.

A.1 I Does the subject of this evaluation increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report? ☐ YES ☒ NO

Bases: The revised UFSAR changes represent a change to the safety analysis and not a plant or procedural change that would introduce an accident initiator that would increase the probability of an accident previously evaluated in the Safety Analysis Report.

II Does the subject of this evaluation increase the consequences of an accident previously evaluated in the Safety Analysis Report? ☐ YES ☒ NO

Bases: The revised analysis presented in the UFSAR changes demonstrates that all acceptance limits are satisfied. Therefore, there is no increase in the consequences of an accident previously evaluated in the Safety Analysis Report.

III Does the subject of this evaluation increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report? ☐ YES ☒ NO

Bases: The revised UFSAR changes represent a change to the safety analysis and not a plant or procedural change that would introduce an accident initiator that would increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report.

IV Does the subject of this evaluation increase the consequences of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report? ☐ YES ☒ NO

Bases: The revised safety analysis presented in the UFSAR changes demonstrates that the acceptance limits of the IVC and RCB structures are not exceeded. The analysis does not impact the Equipment Qualification in these subcompartments. No other components, systems, or structures are impacted by this analysis. Therefore, there is no increase in the consequences of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report.

A.2 I Does the subject of the evaluation create the possibility of an accident of a different type than any previously evaluated in the Safety Analysis Report? ☐ YES ☒ NO

Bases: The revised analysis presented in the UFSAR changes demonstrates that all acceptance limits are satisfied for an accident already addressed in the UFSAR. Therefore, the change does not increase the possibility of an accident of a different type than any previously evaluated in the Safety Analysis Report.

II Does the subject of this evaluation create the possibility of an different type of malfunction than any previously evaluated in the Safety Analysis Report? ☐ YES ☒ NO

Bases: See response to A.2.I

A.3 I Does the subject of this evaluation reduce the margin of safety as defined in the basis for any Technical Specifications? ☐ YES ☒ NO

Bases: The revised analysis presented in the UFSAR change demonstrates that all acceptance limits are satisfied. Therefore, there is not a reduction in the margin of safety as defined in the basis for any Technical specifications.

B. 1. X ________ All of the above questions were answered NO; therefore, the originating document does not involve an Unreviewed Safety Question.

2. ________ One or more of the above questions was marked YES; therefore, the originating document involves an Unreviewed Safety Question. The originating document, as presented, shall NOT be implemented without prior approval by the NRC. Provide a recommendation for disposition of the Unreviewed Safety Question below. Refer to IP-1.19Q for processing licensing amendments. Further processing of this form to the PORC, Plant Manager and NSRB is not required.

RECOMMENDED DISPOSITION: Approve the USQE and Assoicate UFSAR Change

PREPARED BY: Charles R. Albury — ORIGINATOR — 3/30/95 Date

APPROVED BY: D C. Lo... — DEPARTMENT MANAGER — 3/31/95 Date

PORC MEETING NO. 95-017 — 3/31/95 Date

APPROVED BY: Lew W Myers — PLANT MANAGER — 3/31/95 Date

REMARKS: JCO 93-004, Revision 5 contants additional infromation concerning this evaluation.

PORC Review Cover Sheet
0PAP01-ZA-0104-1
(Page 1 of 1)
(Sample)

Originating Document No. JCO 93-0004 Revision No. 5

TITLE MSLB Blowdown Model for IVC

The PORC has reviewed this item and has determined that (check as appropriate):

It ___ does ✓ does NOT involve an UNREVIEWED SAFETY QUESTION.

It ___ does ✓ does NOT adversely impact plant nuclear safety.

It ___ does ✓ does NOT adversely impact the health and safety of plant personnel or the public.

It ✓ does ___ does NOT require further review by the Plant Mgr, the NSRB, or other individuals/groups.

✓ Plant Manager Unit 1 2 ✓ NSRB ___ other (specify below)

REMARKS

The PORC recommends this item for:

✓ APPROVAL ___ DISAPPROVAL ___ OTHER ___ PORC MEETING NO. 95-017

Completed by [signature] DATE 3/31/95
PORC Secretary

This form, when completed, **SHALL** be retained in accordance with the retention requirements of the originating document.

JCO APPROVAL COVER SHEET (TYPICAL)
0PGP05-ZN-0005-01
(Page 1 of 1)

JCO Approval Cover Sheet

Initiation Date:
Expiration Date; (Unit 1) N/A
(Unit 2) N/A

Subject: MSLB BLOWDOWN MODEL FOR IVC

JCO No. 93-0004 Revision No. 5

Applicable Units: 1 & 2

Summary: This JCO is no longer required and is being closed. The design and licensing basis has been revised to delete the need for this JCO

Prepared by/Date: D. G. Lee / 3/31/95
Organization, Manager Responsible for JCO — Date

Concurrence With/Date: [signature] / 3/31/95
General Manager, Nuclear Licensing NUCLEAR/ASSURANCE — Date

Recommended by
PORC Meeting No./Date: /

Approved/Date: [signature] / 3/31/95
Plant Manager, Unit 1 — Date

Approved/Date: W. [signature] / 3/31/95
Plant Manager, Unit 2 — Date

Affected Structure, System, and Components (SSCs)

Unit 1: IVC STRUCTURE
RCB STRUCTURE

Unit 2: SAME AS UNIT 1

References:

Unit 1:
SPR 93-2415
LCTS
IEP 93-55

Unit 2:
SPR SAME AS UNIT 1
LCTS

This Form, when completed, SHALL be retained for the life of the plant.

TABLE OF CONTENTS

REASON FOR REVISION

The purpose of Revision 5 is to close this JCO. The JCO is being closed because the information contained in Licensing Change Notice 1962 and USQE 95-0013 shows that the MSLB issue does not result in an Unreviewed Safety Question. No further actions are required for this JCO.

EXPIRATION DATE

Not applicable for closure.

1.0 IDENTIFICATION

SPR 93-2415 was originated because of concern that the Main Steam Line Break (MSLB) blowdown model for the Isolation Valve Cubicle (IVC) does not account for moisture carry-over from the steam generators; therefore, the blowdown may be non-conservative. This issue was discovered during a review of NRC Information Notice 93-55, "Potential Problems with Main Steamline Break Analysis for Main Steam Vaults/Tunnel." Note that this issue does not affect the Equipment Qualification (EQ) temperature analysis for equipment located within the IVC.

As part of the investigation of SPR 93-2415, Plant Analysis has identified that the mass and energy releases used in other subcompartment pressure/temperature (P/T) analyses are not conservative. The following calculations were reviewed and found to be impacted:

- NC-7012, "IVC P/T Analysis MSLB,"
- NC-7023, "Main Feedwater Line Break P/T Analysis in IVC,"
- NC-7028, "MSLB Subcompartment Analysis," and
- NC-7048, "Main Feedwater Line Break Subcompartment Analysis in RCB."

In particular, the mass and energy releases used in these analyses did not consider the hot zero power conditions with the MSIVs open. The following non-conservative assumptions were used in the IVC P/T analysis (NC-7012):

- the initial steam generator pressure was assumed for full power conditions (1100 psia); the limiting initial pressure would occur at zero power conditions (1266 psia including instrument uncertainties); and
- only steam release from the affected steam generator was considered; backflow steam from the other three steam generators and the main steam piping was not considered.

This JCO is being performed in accordance with STP procedure 0PGP05-ZN-0005 "Justification for Continued Operation," Revision 0, 3-11-94.

2.0 OPERATION

The four main steam lines passing through the IVC structures are housed in separate cubicles isolated from one another by two foot thick reinforced concrete walls. The main function of the structure is to isolate the main steam lines from one another so that a break in one steam line cannot affect adjacent main steam lines.

Immediately following a postulated main steam line break in the IVC, a very rapid pressurization of the affected IVC cubicle would occur. The pressure buildup would be limited by the blowout panels located at the top of the IVC. A ΔP of 0.69 ± 0.14 psid causes the IVC blowout panels to give way and limit pressure buildup. However, due to the dynamic nature of the event, the blowout panels cannot relieve pressure quickly enough to avoid a pressure spike substantially in excess of 0.69 psi during the first second following the postulated rupture. Preliminary analysis for this JCO shows that the pressure would peak at 13.1 psid.

The RCB operates as a passive structure which protects the safety-related equipment within it.

3.0 SAFETY FUNCTION

- 10CFR50, Appendix A, General Design Criterion (GDC) 4, *Environmental and dynamic effects design bases*, discusses the essential safety and regulatory criteria of the IVC and the RCB. GDC 4 states:

  "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

- Section 3.6.1, "Plant Design for Protection Against Postulated Piping Failure in Fluid Systems Outside Containment," of the STP Safety Evaluation Report (SER), and its supplements, also discusses safety and regulatory criteria to be met by the IVC and its subcompartments.

- Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," of the STP SER, and its supplements, discusses safety and regulatory criteria to be met by the RCB and its subcompartments.

- Section 6.2.1.2, "Subcompartment Analysis," of the STP SER, and its supplements, discusses safety and regulatory criteria to be met by various RCB subcompartments.

The following criteria were used or considered in the various analyses discussed in Section 4.0 of this JCO:

- ANSI/ANS-56.10-1982, "Subcompartment Pressure and Temperature Transient Analysis in Light Water Reactors," provides guidance for subcompartment analyses including short-term pressure and temperature transients to which subcompartments will be exposed as a results of postulated line breaks. Also, determination of long-term pressure and temperature transients resulting from both normal and abnormal occurrences are discussed. The standard considers subcompartments located both inside and outside of containment.

- ANSI/ANS-58.2-1980 & 1988, "Design Basis For Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," provides guidance on the design basis for protection of light water reactors from the following potentially adverse effects of a postulated pipe rupture: pipe whip, pipe internal loads, jet impingement, compartment pressurization, environmental conditions, and flooding.

- 10CFR100, Appendix A, Section VI.(a)(1), discusses engineering design as it applies to a Safe Shutdown Earthquake. Specifically, it is stated that "[I]t is permissible to design for strain limits in excess of yield strain in some of these safety-related structures, systems, and components during the Safe Shutdown Earthquake and under the postulated concurrent conditions, provided that the necessary safety functions are maintained."

- General Civil/Structural Design Criteria 5A360SQ1001 governs all structural analysis at STP. Table 11 of the Criteria defines the required design load combinations and load amplification factors.

4.0 SAFETY ANALYSIS

Plant Analysis and DED performed several analyses and evaluations as a result of this issue. The following actions were completed for this Safety Analysis:

- twenty Nuclear Calculations and one Civil Calculation were reviewed for impact;
- pressure/temperature analyses were performed for the IVC and the RCB;
- the existing structural analysis for the IVC was reviewed to determine the impact of the revised pressures;
- short-term and long-term main steam line break evaluations and analyses were made for the IVC and the RCB;
- feedwater line break evaluations were made for the IVC and the RCB; and
- equipment qualification pressure/temperature evaluations were performed for the IVC and the RCB.

4.1 Calculations Reviewed

The following calculations were reviewed with respect to this analysis and found to not be impacted:

- NC-7007, "MSLB - Containment Pressure / Temperature Analysis,"
- NC-7017, "Mass/Energy Release From SGBD Line Break in IVC,"
- NC-7018, "AFW Turbine Driven Pump Steam Supply P/T Analysis,"
- NC-7019, "AFW Line Break P/T Analysis in IVC,"
- NC-7020, "SGBD - P/T Analysis in IVC,"
- NC-7030, "MSLB Forcing Function Analysis for MS1001/MS1002,"
- NC-7038, "Forcing Functions Due to a Spectrum of FWLBs,"
- NC-7042, "MSLB Forcing Function Analysis for MS1003/MS1004,"
- NC-7044, "IVC EQ MSLB,"
- NC-7047, "MSLB - Containment Pressure / Temperature Analysis for Split Breaks,"
- NC-7055, "Inflatable Seal Equipment Qualification Temperature,"
- NC-7057, "IVC P/T Analysis in Five-way Restraint Area,"

- NC-7061, "IVC Corridor P/T Analysis at El. 10' Due to SGBD Line,"
- NC-7062, "IVC Pressurization Following a Break in AFW System at Low Power,"
- NC-7063, "Electrical Equipment Qualification Inside Containment," and
- NC-7064, "Post-Accident Polar Crane Temperatures."

The following calculations were reviewed with respect to this analysis and found to be impacted:

- NC-7012, "IVC P/T Analysis MSLB,"
- NC-7023, "Main Feedwater Line Break P/T Analysis in IVC,"
- NC-7028, "MSLB Subcompartment Analysis,"
- NC-7048, "Main Feedwater Line Break Subcompartment Analysis in RCB," and
- CC-6251, "IVC Reanalysis."

4.2 Pressure/Temperature Analysis

Thermal Hydraulics performed analysis to determine the impact of the revised assumptions on the IVC and RCB P/T analysis. The analysis was performed using the RETRAN-03 computer code to determine the mass and energy releases which were then used by the GOTHIC computer code to determine the pressure response of the IVC & RCB subcompartments. RETRAN-03 is a RELAP4-type code and meets the criteria specified by ANSI/ANS 56.10-1982 and SER Section 6.2.1.2. The GOTHIC models were benchmarked against the results from Bechtel's COPDA models with acceptable results. The computer models were revised as necessary to conform with the applicable portions of ANSI/ANS-58.2-1988, "Design Basis For Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture" and ANSI/ANS 56.10-1982, "Subcompartment Pressure and Temperature Transient Analysis in Light Water Reactors." The revised analysis makes the following assumptions which are significantly different from those used in the original analysis:

- the initial pressure is at 1266 psia which is the saturation pressure at the zero power temperature of 567°F (plus 7°F for instrument uncertainty); the original analysis used the full power pressure of 1100 psia;
- the RETRAN model includes all steam sources including the three unaffected steam generators and the steam headers; the original Bechtel model did not include all steam sources; and
- the impact of steam moisture carry-over was considered.

The pressure response of the subcompartments is divided into two distinct phases, the short-term response and the long-term response. The short-term response identifies the initial peak pressure used in the dynamic pressure loading analysis of surrounding structures. Short-term analysis includes approximately the first 0.5 seconds of the transient. The fluid phase of the short-term portion of the transient is 100% quality steam because the moisture carry-over from the steam generators does not reach the break location until after the initial pressure peak has been experienced.

The second phase (long-term response) identifies the pressure response associated with the moisture carry-over from the steam generator. For the purposes of these analyses, a 4% quality steam was assumed as discussed in ANSI/ANS standard 58.2-1980, Appendix E. (This is the ANSI standard referenced in IEN 93-55.)

4.3 Structural Analysis of IVC Main Steam Line Break and Results of IVC Short-Term Evaluation

Using the results of the short-term IVC pressure analysis, DED evaluated the impact on the structural analysis and design of the IVC contained in calculation CC-6251. The existing analysis is based on calculated pressure peaks at nine different locations (nodes) in the IVC, amplified by appropriate load factors (as required by General Structural Design Criteria SQ1001).

In comparing the calculated new pressure peaks to the nine peaks used in the current analysis, the maximum pressure in the limiting node (the break node) for the IVC decreases from 14.0 psid to 11.5 psid. However, the maximum building stress due to dynamically applied pressure loads depends on the shape of the pressure versus time function, not just on the magnitude of the peak pressure. This effect is incorporated by use of "dynamic load factors" to amplify the pressure peaks. New dynamic load factors, which are higher than the ones used in the current analysis, have been calculated by Sargent & Lundy. Using the new dynamic loading factors, it is possible that one local region of the IVC may be slightly overstressed by as much as 10%, however structural integrity of the IVC will not be affected for the following reasons:

- if a localized area is overstressed by 10% in this concrete configuration, the stresses will be redistributed, and thus the loads would be carried by other structural areas or members;

- the IVC building design does not consider in-situ material strengths. The design strength of the concrete (f_c') is 4000 psi whereas actual strength of the concrete is in excess of 6000 psi. Actual strength of Grade 60 reinforcing steel is typically 5 to 10% above the minimum required 60 ksi.

Unlike the IVC break node, other IVC nodes experienced increases in peak pressure. However, based on a review of the existing analysis, DED has concluded that the maximum stress in the building is strongly correlated to the maximum pressure in the break node, and the fact that the non-peak pressures have increased is of relatively little significance.

Accordingly, the IVC structure will maintain its overall structural integrity as a result of the new pipe rupture loads. Therefore, the revised P/T analysis does not affect continued safe operation.

4.4 Results of RCB Short-Term Main Steam Line Break Evaluation

Results of the pressure analysis on the RCB short-term main steam line break show that the new calculated peak pressures are slightly higher than those originally calculated by Bechtel. This small increase has been evaluated and found to be within the existing margin of the structure. Therefore, there is no adverse impact on the RCB structural analysis resulting from a short-term main steam line break.

4.5 Results of IVC & RCB Long-Term Main Steam Line Break Analysis

As discussed in IEN 93-55, moisture carry-over from the steam generators may result in higher peak pressures in the subcompartments. Due to the high pressure in the main steam lines, choked flow would occur in the event of a postulated break. Under choked flow conditions, moisture in the steam lines would decrease the enthalpy of the break fluid but increase the break flow rate. The overall effect is an increase in the energy released from the break because the decrease in enthalpy cannot compete with the increase in break flow rate. The analysis of the long-term response with moisture carry-over for the limiting subcompartments shows that the calculated pressures in the IVC & RCB exceeded the peak pressure calculated for the short-term analyses.

The following assumptions were used for the long-term analyses using moisture carry-over:

- steam generator pressure was held constant at 1266 psia, with no credit taken for depressurization of the steam generators;
- The impacted steam generator depletes its water mass approximately 23 seconds after the break occurs. Calculation of steam generator water mass deplection time assumes the following: (1) the full mass of water in the steam generator at hot zero power plus 10% for uncertainties, (2) AFW flow is added coincidnet with the break at the runout flow of 1210 GPM, and (3) 4% quality steam;
- After the mass in the effected steam generator is depleted, the mass and energy release from this generator is significantly reduced. At this time, the MSIVs are also assumed to close. This is conservative because the a MSIV closure is expected to occur at approximately 15 seconds based on a low steam line pressure signal; and
- The mass flow rate addition from AFW is not sufficient to produce moisture carry-over. This is based on the fact that the AFW pump runout flow of 160 lbm/sec is significantly less than the main steam mass flow rate of 1175 lbm/sec at full power conditions. At full power conditions, moisture carry-over is not an issue.

The long-term effects of the IVC pressure response for the limiting subcompartment (Node 7) is less than the peak pressure for the Bechtel analysis. The pressure reponse has been evaluated by DED and found acceptable. For the RCB, the peak pressure differential between the main containment subcompartment and the break subcompartment is less for the long-term response than the short-term response. Therefore, the long-term effects of a main steam line break in the IVC and RCB are bounded by the short-term effects.

4.6 Results of IVC & RCB Main Feedwater Line Break Evaluation

The main feedwater line break analyses for the IVC (NC-7023) and the RCB (NC-7048) were reviewed. All sources of feedwater were considered in the analyses, but full power conditions were assumed. Assuming full power conditions serves to increase the enthalpy of the fluid considered in the break yet decrease the initial assumed pressure. Although the sensitivity of these effects were not evaluated by Bechtel, the results of the current pressure analyses show

that the feedwater line break is bounded by the main steam line break. The peak pressures for the full power cases are shown below.

Subcompartment	Limiting Feedwater Line Break Peak Differential Pressure (Current Analysis)	Limiting Main Steam Line Break Peak Differential Pressure (Current Analysis)
IVC	4.3 psid (Node 11)	14.0 psid (Node 7)
RCB	1.2 psid (Node 7)	13.7 psid (Node 3)

4.7 Results of IVC & RCB EQ P/T Analysis Evaluation

A review of the EQ P/T analyses for the IVC (NC-7044) and the RCB (NCs-7007, 7047, 7055, 7063, and 7064) show that the mass and energy releases calculated by Westinghouse using the LOFTRAN code properly assumed steam from all sources and at the correct initial pressures. If moisture carry-over were considered for EQ analyses, the subcompartment temperatures would be reduced. Therefore, the EQ P/T analyses remain bounding.

5.0 REQUIRED COMPENSATORY ACTIONS

There are no compensatory actions required as a result of this JCO.

6.0 CORRECTIVE ACTION

1. Upon receipt of the final pressure/temperature analyses from Nuclear Fuel and Analysis, DED Civil/Structural Group will reevaluate the IVC structure for the effects of the pressure loading.

2. Nuclear Fuel and Analysis and DED will incorporate the analyses and evaluations discussed in Section 4.0 into the STP design basis.

These actions were completed on March 30, 1995. Completion of these corrective actions removed the necessity for this JCO.

7.0 REPORTING

This section addresses the need for reporting under 10CFR50.72 and 10CFR50.73 as well as potential 10CFR21 concerns.

10CFR50.72: There are no required reportable concerns pursuant to 10CFR50.72.

10CFR50.73: The IVC structure will maintain its overall structural integrity as a result of the new pipe rupture loads. 10CFR50.73(a)(2)(ii) applies, but no structural system will be degraded or in an unanalyzed condition as a result of the new pipe rupture pressure loading provided. Therefore, this item is not reportable.

10CFR21: There are no known 10CFR21 concerns as a result of this JCO.

8.0 REFERENCES

1. 9A010SS1009; "Specification for Controlled Release Roofing for the Isolation Valve Cubicle;" Revision 4.
2. Calculation RC-6548; "Pipe Stress Analysis Calculation"; Revision 6.
3. NUREG-0781; "Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 and 2;" April 1986.
4. NUREG-0781; "Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 and 2;" Supplement 1; September 1986.
5. Station Problem Report 93-2415.
6. NRC Information Notice 93-55; "Potential Problems with Main Steamline Break Analysis for Main Steam Vaults/Tunnel."
7. Calculation CC-6251; "IVC Reanalysis;" Revision 3.
8. ANSI/ANS-56.10-1982, "Subcompartment Pressure and Temperature Transient Analysis in Light Water Reactors."
9. ANSI/ANS 58.2-1980, "Design Basis For Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture."
10. ANSI/ANS 58.2-1988, "Design Basis For Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture."
11. Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants."
12. 5A360SQ1001 General Civil/Structural Design Criteria.
13. B&R TRD 2A700GP003-C; Show Cause Report on Concrete Structures; "Review of Safety-Related Concrete Structures Including Embedments."

14. NUREG-0781, "Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 and 2.

ATTACHMENT - A
JCO OPERATIONAL IMPACT STATEMENT

JCO No. 93-0004, Revision 3
JCO Title: MSLB Blowdown Model for IVC

A. To maintain the validity of this JCO, Plant Operations must do the following:

   1. There are no actions required of Plant Operations in order to maintain the validity of this JCO.

Change Number 1962 Date 3/29/95

Originator Charles Albury Dept NPLA

Change Description Main Steam Line Break in IVC

Initiating Documentation JCO 93-004

USQE Number 95-0013

Unit(s) Affected: Unit 1 ✓ Unit 2 ✓

Implementation Status: Unit 1 Completion Date ______

Unit 2 Completion Date ______

Reviewed and Approved by ______
Supervising Engineer, Nuclear Licensing Date

Reviewed and Approved by (ER, UFSAR 2.1, 2.2, 2.3) ______
Manager, Technical Services Department Date

Reviewed by (OQAP changes only) ______
General Manager, Nuclear Assurance Date

This form when completed, shall be retained for the life of the plant.

APPENDIX 3.6.A

ISOLATION VALVE CUBICLE SUBCOMPARTMENT ANALYSIS

3.6.A.1 Design Features. The Isolation Valve Cubicle (IVC) is located between the Containment and Turbine Generator Buildings (TGBs) on the north side of the Containment. Figures 1.2-21 through 1.2-25 provide the plan and elevation views of this area. The IVC consists of four cubicles with each cubicle designed to accommodate equipment and piping pertaining to each of the four trains of the steam feedwater system, thus meeting the train separation criteria.

At lower levels (between El. 10 ft-0 in. and 34 ft-0 in.) each train has an auxiliary feedwater (AFW) pump. Three of the pumps are motor-driven while the fourth is turbine-driven. Watertight doors assure the separability of the auxiliary pump cubicles from one another in the event of flooding of any one of the cubicles due to a pipe break. Main steam (MS) and main feedwater (FW) pipes run through the IVC above El. 34 ft-0 in. extending from the Containment penetrations to the five-way bending-torsional restraints mounted between two walls on the north end of the IVC. The main steam isolation valves (MSIVs), MS safety valves, main feedwater isolation valve (MFIV), etc. are located in this compartment. A sloped metal roof covers the top of the IVC. The roof will lift off the affected cubicle in the event of a pressure build-up due to a pipe break in one of the cubicles. The AFW pump cubicles relieve their pressure build-up in the event of a AFW pipe break through the opening at El. 34 ft-0 in. from whence it is eventually vented to the atmosphere via the roof in the IVC.

3.6.A.2 Design Evaluation. ~~The subcompartment pressure transients were determined using the COPDA computer code. Details of the code are given in Section 6.2.1.2.3.~~ The MS and FW piping in this compartment is designed to the break exclusion criteria, stated in Section 3.6.2.1, for those portions of the piping passing through the primary containment and extending to the first pipe whip restraint past the first outside isolation valve. Accordingly, mechanistic pipe breaks are not postulated in the MSIV/MFIV piping. However, to provide an additional level of assurance of operability of safety-related equipment in this compartment, the building structures and safety-related equipment are designed to environmental conditions (pressure temperature and flooding) that would result from a break, equal to one cross-sectional area of the MS and main FW piping. Adequate venting is provided to limit the pressurization of the cubicles to below the design pressures of the wall.

The following cases were analyzed to determine the worst environmental conditions for the IVC.

1. Main steam line break (MSLB) equivalent to the area of a single area rupture

2. Main FW line break due to a single area break

3. AFW line double-ended break in the AFW cubicle

4. Double-ended steam generator (SG) blowdown line break in common corridor area

In general the calculated maximum pressures resulting from an MSLB are greater than those calculated for the other postulated break types. There is one exception. A break in the SG blowdown line results in the highest pressure calculated for the common corridor area north of the AFW pump cubicles at El. 10 ft. 2

Insert (A)

~~The RELAP5 computer code (Ref. 6.2.1.2.6) has been used to calculate the short-term blowdown of the main steam line. Results are presented in Table 3.6.A-1.~~

Short-term ~~blowdown data for the AFW line break and the SG blowdown line break~~ were calculated using the methodology of References 3.6-9, 3.6.A-6 and 3.6.A-7. for the main FW line break and AFW line break

The COPDA nodalization scheme ~~selected~~ for the IVC model is shown in Figure 3.6.A-1. The common corridor area is not part of this model. The nodal boundaries have been selected wherever there are flow restrictions (such as grating platforms). As mentioned, the roof of the IVC is covered by built-up metal panels. The differential pressures at which these panels lift is 0.8 psig. The weight of these panels is 3 lbs/ft². The panel is assumed to move parallel to its original position (note the panel has a small slope away from the Containment Building) till it clears the sidewalls of the IVC. Once the panels clear the walls, it is assumed to lift away from the path of the flow of the steam-air mixture to the atmosphere. Thus, this movement of the panels above its nominal position creates movable nodes 10 and 11 shown in Figure 3.6.A-1A. The node volume and junction parameters of the IVC are given on Table 3.6.A-2A. Node 10 and 11 have variable properties as the panel moves above its nominal position. The vent area and the volume of these nodes are given in Tables 3.6.A-3 and 3.6.A-4.

Insert (B)

The nodalization model selected for the common corridor area is shown in Figure 3.6.A-5. The node and junction parameters of the common corridor area of the IVC are given in Table 3.6.A-6.

Results of the cases which yield maximum pressures in the various nodes of an IVC cubicle including the associated AFW pump room are presented in Figure 3.6.A-3. In MSLB case 1 ~~all blowdown~~ is assigned to node 6, while in MSLB case 2 ~~all blowdown~~ goes to node 7. In the AFW break case ~~all blowdown~~ is assigned to node 2. ~~The peak~~ the mass and energy release pressures for the limiting case in each node are indicated in Table 3.6.A-2.

Peak pressures for the SG blowdown line break in the common corridor area of the IVC structure are presented in Table 3.6.A-6.

For generating the equipment qualification temperatures and pressures of the IVC a simpler 3-node model of the IVC has been used and the volume and junction properties were inputted into a modified COPDA code named FLUD (see Section 3.6.A.3 for discussion of FLUD). The simplified model consists of 3-nodes with node 1 being the auxiliary pump room between El. 10 ft and 32 ft, node 2 is between El. 34 ft and 55 ft-6 in. and node 3 occupying space above 55.5 ft. Out of the various cases considered, MSLB produced the limiting temperatures and pressures in the IVC. The long term ~~blowdown~~ mass and energy release used in the analysis is presented in Table 3.6.A-5 and the temperature profiles are given

INSERT A

The pressure analysis for the Main FW line break, AFW line break, and steam generator blow down line break were modeled using the COPDA computer code. Details of the code are given in section 6.2.1.2.3. Short term mass and energy releases were calculated using the methodology of References 3.6-9, 3.6.A-6, and 3.6.A-7.

The MSLB subcompartment pressure analysis was determined using the GOTHIC 4.0 HLP-001 (Reference 3.6.A-8) computer code. Details of the code are given in Section 3.6.A.6. The short term mass and energy releases for the IVC subcompartment pressure analysis were determined using the RETRAN-03 computer code, which is discussed in Section 6.2.1.4.7.

INSERT B

For the MSLB analysis, the nodalization scheme is presented in Figure 3.6.A.-1B. The nodalization is similar to that of the main feedwater line and AFW line break with the exception of the modeling of the movable panels. The position of the movable panels are modeled as a function of IVC pressure using the STEM_TRAVEL code (Reference 3.6.A-9). The node and junction parameters of the IVC are given on Table 3.6.A-2B.

in Figure 3.6.A-4. The mass and energy release ~~Blowdown~~ has been obtained using Westinghouse LOFTRAN code (Ref. 3.6.A-5).

3.6.A.3 FLUD, A Compartment Differential Pressure Analysis Code. This describes the computational procedure and the analytical techniques used in FLUD. The analytical basis for COPDA is described in Reference 6.2.1.2-2. The set-up of initial conditions, the determination of the thermodynamic state point at subsequent time increments, and computation of energy and mass transport between one time step is discussed in Sections 3.6.A.3.1, 3.6.A.3.2, and 3.6.A.3.3 for FLUD. Selection was made of the control volume and flow path configuration that resulted in the best representation of the pressure transients in the compartments along the flow paths from the break. The major differences between FLUD and COPDA (Ref. 3.6.A-4) are the use of steam table curve fits (Section 3.6.A.3) instead of table look-ups, the equation of state which is a first-order virial expansion (discussed in Section 3.6.A.3.1) and the capability of wall heat transfer calculation. The fluid flow equations (compressible equations, HEM model and integrated momentum equation) used in COPDA have been reproduced in the FLUD code. It may be observed from the FLUD flowchart in Figure 3.6.A-2 that the calculational procedures for FLUD and COPDA are very similar.

3.6.A.3.1 Equation of State - This section describes how FLUD determines the thermodynamic state for each compartment in a system of interconnected compartments.

The thermodynamic system (compartment) is assumed to be in equilibrium. The states assumed by the air-steam-water mixture can be described in terms of thermodynamic coordinates, P, V, and T referring to the mixture as a whole. The equation of state is derived from a first order virial expansion as presented in Reference 3.6.A-1. Using the molecular theory of gases, the following equation of state for an air-steam mixture is obtained assuming negligible air-steam molecular interaction:

$$P = (M_a R_a + M_s R_s) \frac{T}{V} + \frac{(M_s)^2}{V} R_s \, T \, B_s(T), \ (\text{lbf/ft}^2) \qquad \text{(Eq. 3.6.A-1)}$$

where the temperature dependence of the second virial coefficient for steam $B_s(T)$ is given by (Ref. 3.6.A-2).

$$B_s(T) = 0.0330 - \frac{75.3137}{T} 10^{3.2658/(T^2 \times 10^{-5} + 1.1308)} \qquad \text{(Eq. 3.6.A-2)}$$

Equation 3.6.A-1 can be rewritten as the sum of the partial pressure of air P_a and the partial pressure of steam P_s where

$$P_a = \frac{M_a R_a T}{V}, \ (\text{lbf/ft}^2) = 0.37043 \frac{T}{v_a}, \ (\text{psia}) \qquad \text{(Eq. 3.6.A-3)}$$

$\dot{M}_{atm}$ - atmospheric mass flow rate

$\dot{M}_{a}$ - mass condensation rate

3.6.A.5 Thermodynamic Properties of Steam, Water, and Air - FLUD uses steam, air, and water properties for various thermodynamic calculations which are performed during each step. The thermodynamic variables needed in FLUD calculations are:

$e_a(T)$ - specific internal energy of air

$P_{sat}(T)$ - saturation pressure of steam

$v_{sat}(T)$ - saturation specific volume of steam

$e_s(T,P)$ - specific internal energy of steam

$v_w(T)$ - specific volume of water

$e_w(T)$ - specific internal energy of water

$T_{sat}(P)$ - saturation temperature of steam

$T_{sat}(v)$ - saturation temperature of steam

$e_{sat}(T)$ - saturation specific internal energy of steam

$h_{sat}(T)$ - saturation specific enthalpy of steam

$h_{fg}(P)$ - enthalpy of vaporization of steam

The "unknown" quantities that can be used to calculate the above nine variables are the macroscopic compartment thermodynamic variables pressure, specific volume, and temperature, P, v, and T, respectively.

The air and water properties $e_a(T)$, $v_w(T)$, and $e_w(T)$ are calculated by fitting polynomials to data in the steam and gas tables (Refs. 3.6.A-2 and 3.6.A-3). The air property $e_a(T)$ was found to be adequately represented by a linear fit. This is no doubt due to the good "ideal gas" behavior of air. Thus,

$$e_a(T) = a_1T \qquad \text{(Eq. 3.6.A-30)}$$

The water properties $v_w(T)$ and $e_w(T)$ and the steam properties $h_{sat}(T)$, $e_o(T)$, and $e_{sat}(T)$ are very nearly straight line functions, but small variations were accomdated by using third order spline polynomial fits of the general form:

$$\text{property }(T) = a_o + a_1T + a_2T^2 + a_3T^3 \qquad \text{(Eq. 3.6.A-31)}$$

For example, for $h_{fg}(P)$:

$$h_{fg}(P) = a_o + a_1P + a_2\,P^2 + a_3P^3{}_3 \qquad \text{(Eq. 3.6.A-33)}$$

The accuracy of the curve fits range between 0.01 percent and 4 percent for the various properties.

Insert C

INSERT C

3.6.A.6 GOTHIC 4.0 HLP-001

GOTHIC 4.0 HLP-001 (Reference 3.6.A-8) is a state-of-the-art program that solves the conservation equations for mass, momentum and energy for multi-component, multi-phase flow. The phase balance equations are coupled by mechanistic models for interface mass, energy and momentum transfer that cover the entire flow regime from bubbly flow to film/drop flow, as well as single phase flows. The interface models allow for the possibility of thermal nonequilibrium between phases, and unequal phase velocities. GOTHIC includes full treatment of the momentum transport terms in multi-dimensional models, with an optional one-dimensional turbulence model for turbulent shear and turbulent mass and energy diffusion. Conservation equations are solved for three fields:

- Steam/gas mixture
- Continuous liquid
- Liquid droplet

The program calculates the relative velocities between the separate but interacting fluid fields, including the effects of two-phase slip on pressure drop. The program also calculates heat transfer between phases, and between surfaces and the fluid. Liquid droplets are transported in the vapor/gas flow. In addition, a mass balance is solved for a solid ice phase. The somewhat simplified model for the ice phase does not provide for transport of ice, and if ice exists within a model, the ice temperature is set to a constant value by code input. Ice remains at its initial temperature throughout a transient. Ice can only change phase to liquid; i.e., there is no direct ice-to-vapor phase change.

The three fluid fields may be in thermal nonequilibrium in the same computational cell. For example, saturated steam may exist in the presence of a superheated pool and sub-cooled drops. The code can model extremely dry noncondensable gases (down to steam partial pressures of 0.001 psia) and has a temperature range from -50 F to 8540 F.

The steam/gas mixture is referred to as the vapor phase and is comprised of steam and, optionally, up to eight different noncondensable gases including air and hydrogen. Mass balances are solved for each component of the steam/gas mixture, thereby providing the volume fraction of each type of gas in the mixture.

Solution of the equations is based upon nodalization of the region of interest, with the principal element of a model being a computational volume. The program features a flexible noding scheme that allows computational volumes to be treated as lumped parameter, one-, two- or three-dimensional, or any combination of these within a single model. As a minimum, a GOTHIC model consists of at least one lumped parameter volume. Subdivision

of a volume into a one-, two- or three-dimensional mesh is based on orthogonal coordinates. Adjacent cells in a subdivided volume communicate through parameters defined by discretization of the governing equations. Separate volumes communicate through what are referred to as junctions or flow paths. A separate set of momentum equations are solved for junctions.

GOTHIC has been verified by HL&P against the applicable portions of ANSI/ANS 56.10-1982 (Reference 3.6.A-10) for subcompartment pressurization analysis.

3.6.A.7 STEM_TRAVEL

STEM_TRAVEL (Reference 3.6.A-9) was developed by HL&P to facilitate the use of GOTHIC in subcompartment pressure/temperature (P/T) transient analysis. The code calculates the panel height corresponding to a given MSLB subcompartment pressure profile generated from the GOTHIC computer code. The vent paths are modeled by an equivalent panel height, and in turn, translated into an equivalent stem travel. GOTHIC generates a new pressure profile based on the new stem travel profile. Iterations between the two codes are done manually until convergence between the stem travel and pressure profiles is obtained. The panel height is obtained from the governing equation for dynamic vent paths as discussed in Appendix E of ANSI/ANS 56.10-1982, "Subcompartment Pressure and Temperature Transient Analysis in Light Water Reactors," Reference 11.

$$M\frac{d^2h}{dt^2} = A_o\left(P_{in} - P_{out} - P_m + \frac{W^2}{\rho A_o^2}\right)$$

where:	M	=	mass of panels
	h	=	current vertical panel displacement,
	A_o	=	fully open vent area
	P_m	=	weight per unit area of the panels,
	P_{in}	=	static pressure at current time in the region beneath the panel,
	t	=	time
	P_{out}	=	static pressure in the region above the panel.
	W	=	mass flow rate
	ρ	=	fluid mass density

The equation is simplified to the following form in STEM_TRAVEL:

$$h = h_o + v_o\tau + \frac{g_c}{m}\left(\frac{1}{6}\frac{P_{in} - P_{in,0}}{\tau}\tau^3 + \frac{P_{in,0} - P_{out} - P_m}{2}\tau^2\right)$$

where h = current vertical panel displacement,

h_o = panel displacement at beginning of time step,

v_o = panel velocity at beginning of time step,

τ = time step size, Δt,

g_c = gravitational constant,

m = mass per unit area of the panels,

P_m = weight per unit area of the panels,

P_{in} = static pressure at current time in the region beneath the panel,

$P_{in,0}$ = static pressure below the panel at beginning of time step, and

P_{out} = static pressure in the region above the panel.

REFERENCES

Appendix 3.6.A:

3.6.A-1 Reif, F.J. Fundamentals of Statistical and Thermal Physics, McGraw-Hill Book Co., p. 183.

3.6.A-2 Kennan, J.H. et al, Steam Tables, John Wiley & Sons, Inc., New York, 1969.

3.6.A-3 Keepan, J.H., and J. Kaye, Gas Tables, John Wiley & Sons, Inc., New York, 1948.

3.6.A-4 Bechtel Topical Report BN-TOP-4 Rev. 1, October 1977, "Subcompartment Pressure and Temperature Transient Analysis". This report was approved by the NRC in February, 1979.

3.6.A-5 Burnett, T.W.T., et. al., "LOFTRAN Code Description", WCAP-7907-P-A (Proprietary Class 2), WCAP-7907-A, (Proprietary Class 3), April 1984.

3.6.A-6 Bilanin, W.J., "General Electric Mark III Pressure Suppression Containment System Analytical Mode", General Electric Topical Report NEDO 20533, June 1974.

3.6.A-7 Sharma, D.F., "Technical Descripton - Annulus Pressurization Load Adequacy Evaluation", General Electric Topical Report NEDO 24548, January 1979.

Insert D

INSERT D

3.6.A-8 George, T.L., et al, "GOTHIC 4.0 HLP-001, Containment Analysis Package," Developed by Numerical Application, Inc. for the Electric Power Research Institute, September, 1993.

3.6.A-9 Futschik, M.W., "STEM_TRAVEL", developed by Houston Lighting and Power Company, February, 1995.

3.6.A-10 ANSI 56.10-1982, "Subcompartment Pressure and Temperature Transient Analysis in Light Water Reactors."

TABLE 3.6.A-1

MAIN STEAM LINE BREAK BLOWDOWN

(An enthalpy of 1200 Btu/lbm is conservatively assumed throughout).

Time (secs)	Steam Generator A or B (lb/sec)
0.0	0.0
0.0066	8700.0
0.01	6090.89
0.013	5763.63
0.025	5531.81
0.05	5227.25
0.10	4890.86
0.125	4972.72
0.15	5190.87
0.20	5036.35
0.25	4899.95
0.30	4809.05
0.35	4763.60
0.40	4718.15
0.45	4718.15
0.50	4672.70

THIS TABLE DELETED

TABLE 3.6.A-2A

IVC SUBCOMPARTMENT NODAL DESCRIPTION

For Main FW Line, AFW Line, and SG Blowdown Line Break

Volume Number	Volume ft^3	Initial Conditions Temp. °F	Pressure psia	Humidity Percent	Flow Path	Flow Area ft^2	Flow Coefficient	L/A ft^{-1}	Calculated Peak Press. psia	Break Type
1.	3588.5	105.0	14.7	30					21.65	MSLB
2.	1977.5	105.0	14.7	30	2 1	75.45	0.78	0.05	21.30	MSLB
3.	5330.95	105.0	14.7	30	3 2	54.16	0.82	0.18	20.62	MSLB
4.	2558.5	105.0	14.7	30	4 3	210.37	0.80	0.024	20.66	MSLB
					4 5	64.60	0.92	0.26		
					4 6	256.84	0.82	0.02		
5.	1453.0	105.0	14.7	30	5 3	115.72	0.80	0.035	23.31	MSLB
					5 7	131.90	0.817	0.04		
6.	2221.7	105.0	14.7	30	6 7	56.52	0.91	0.27	22.99	MSLB
					6 8	195.42	0.80	0.05		
7.	1262.26	105.0	14.7	30	7 9	90.29	0.79	0.09	28.87	MSLB
8.	7957.44	105.0	14.7	30	8 9	80.53	0.85	0.07	19.83	MSLB
					8 10	172.5	0.65	0.038		
9.	5448.47	105.0	14.7	30	9 11	178.	0.72	0.04	19.26	MSLB
					9 12	15.5	0.54	0.48		
10.	(Please see Table 3.6.A-3 for details for this node.)									
11.	(Please see Table 3.6.A-4 for details for this node.)									
12.	1.0E22	105.0	14.7	30						

2

TABLE 3.6.A-2 B

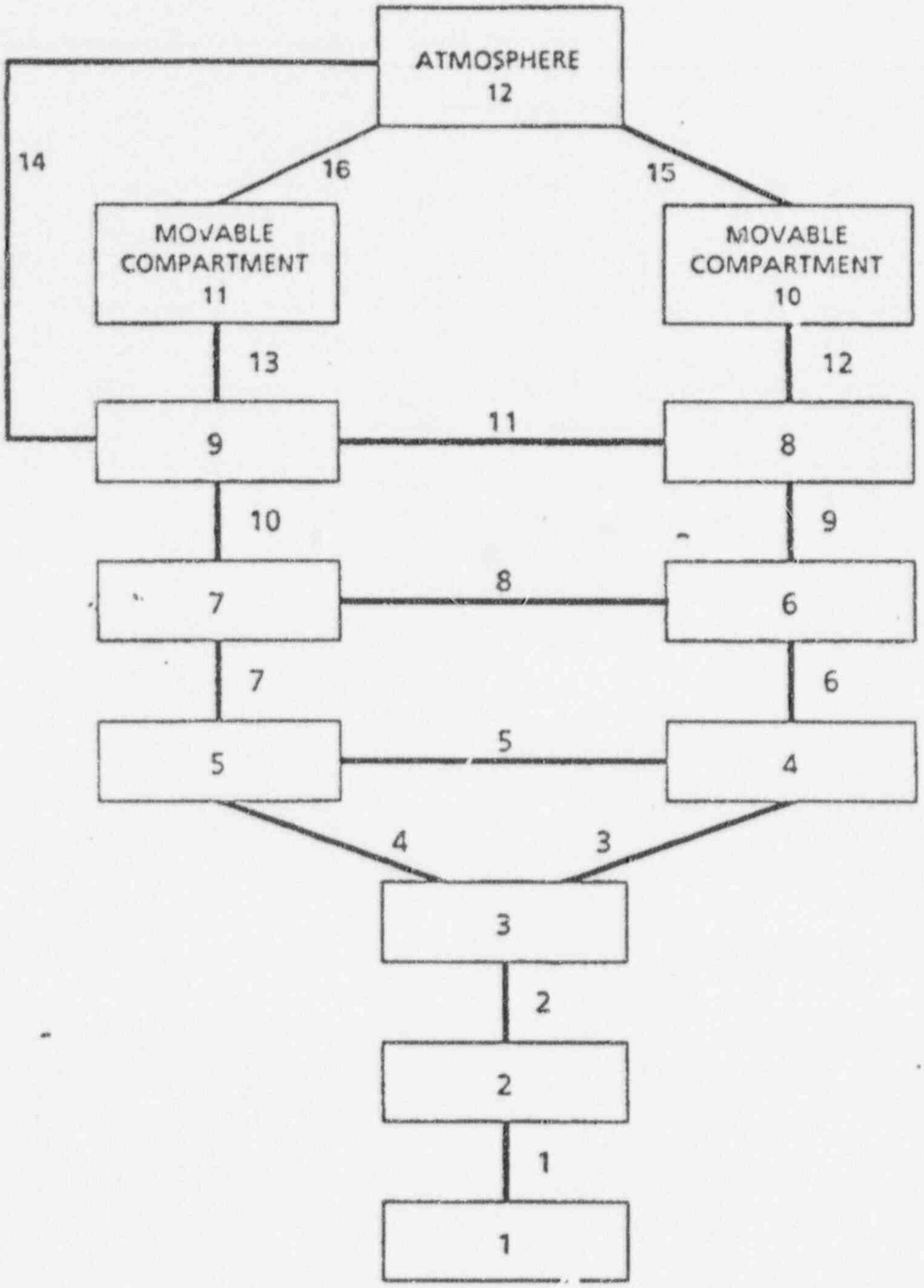
IVC SUBCOMPARTMENT NODAL DESCRIPTION

For Main Steam Line Break

Volume Number	Volume ft^3	Initial Conditions: Temp. °F	Initial Conditions: Pressure psia	Initial Conditions: Humidity Percent	Flow Path	Flow Area ft^2	Flow Coefficient	L/A ft^{-1}	Calculated Peak Press. psia	Break Type
1.	3588.5	~~105.0~~ 104.0	14.7	~~30~~ 20					21.65	MSLB
2.	1977.5	~~105.0~~ 104.0	14.7	~~30~~ 20	2 1	75.45	~~0.78~~ 0.79	0.05	21.30	MSLB
3.	5330.95	~~105.0~~ 104.0	14.7	~~30~~ 20	3 2	54.16	~~0.82~~ 0.83	0.18	20.62	MSLB
4.	2558.5	~~105.0~~ 104.0	14.7	~~30~~ 20	4 3	210.37	~~0.80~~ 0.81	0.024	20.66	MSLB
					4 5	~~64.60~~ 50.32	~~0.92~~ 0.92	0.26		
					4 6	256.04	0.82	0.02		
5.	1453.0	~~105.0~~ 104.0	14.7	~~30~~ 20	5 3	115.72	~~0.80~~ 0.81	0.035	23.31	MSLB
					5 7	~~131.90~~ 115.42	~~0.817~~ 0.81	0.04		
6.	2221.7	~~105.0~~ 104.0	14.7	~~30~~ 20	6 7	~~56.32~~ 35.47	~~0.91~~ 0.98	~~0.27~~ 0.28	22.99	MSLB
					6 8	195.42	0.80	0.05		
7.	1262.26	~~105.0~~ 104.0	14.7	~~30~~ 20	7 9	90.29	0.79	0.09	28.87	MSLB
8.	~~7952.66~~ 8131.36	~~105.0~~ 104.0	14.7	~~30~~ 20	8 9	~~80.53~~ 101.59	~~0.85~~ 0.965	~~0.07~~ 0.077	19.83	MSLB
					8 ~~10~~ 3P	172.5	0.65	0.038		
9.	~~5448.47~~ 5606.85	~~105.0~~ 104.0	14.7	~~30~~ 20	9 ~~11~~ 2P	178.	0.72	0.04	19.26	MSLB
					9 ~~12~~ 1P	15.5	0.54	0.48		
~~10.~~ 3P	~~(Please see Table 3.6.A-3 for details for this node.)~~ NA	104.0	14.7	NA						
~~11.~~ 2P	~~(Please see Table 3.6.A-4 for details for this node.)~~ NA	104.0	14.7	NA						
~~12.~~ 1P	~~1.0E22~~ NA	~~105.0~~ 104.0	~~14.7~~ 14.7	~~30~~ NA						

Pressure Boundary conditions

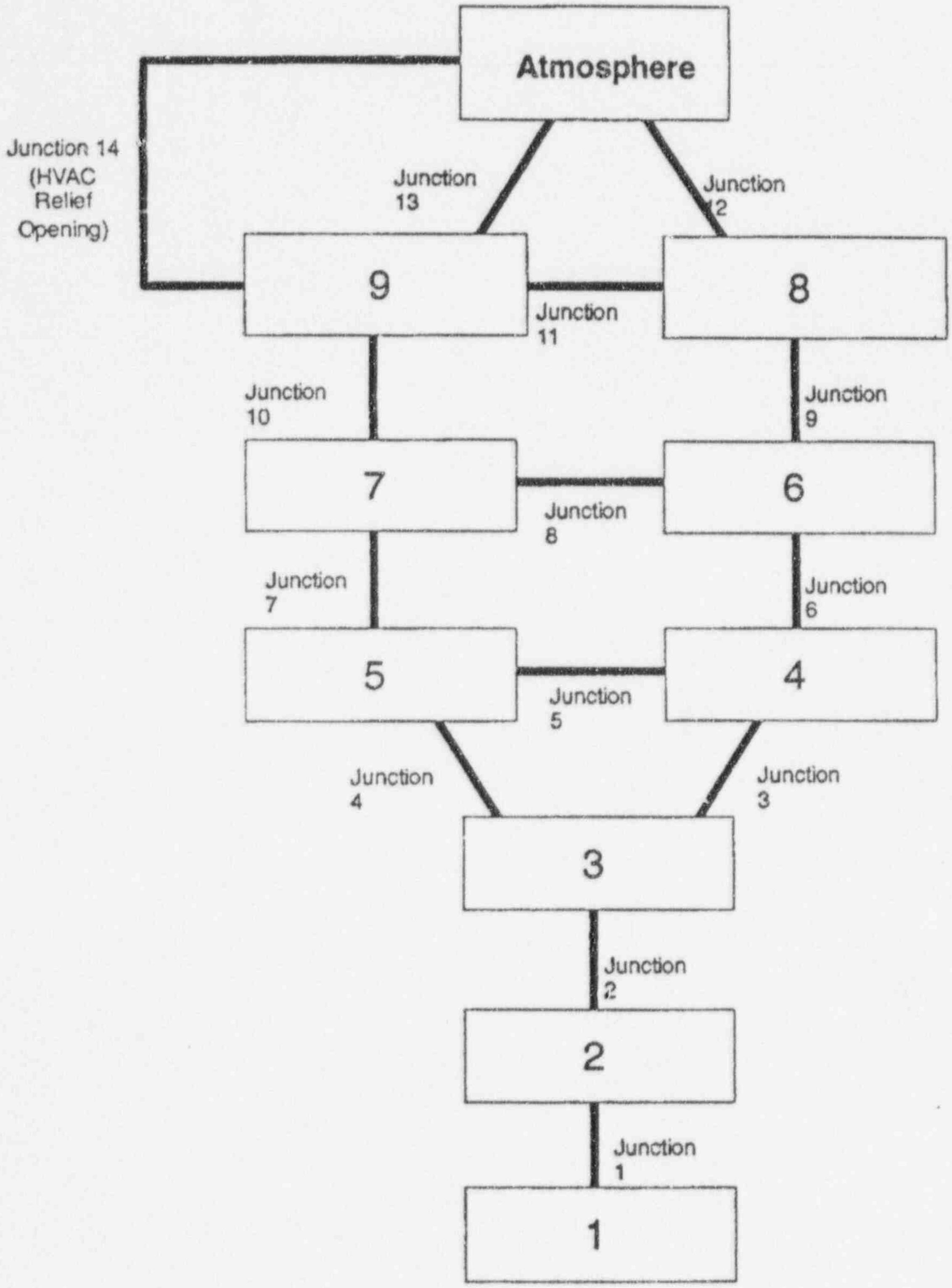
2



SOUTH TEXAS PROJECT
UNITS 1 & 2

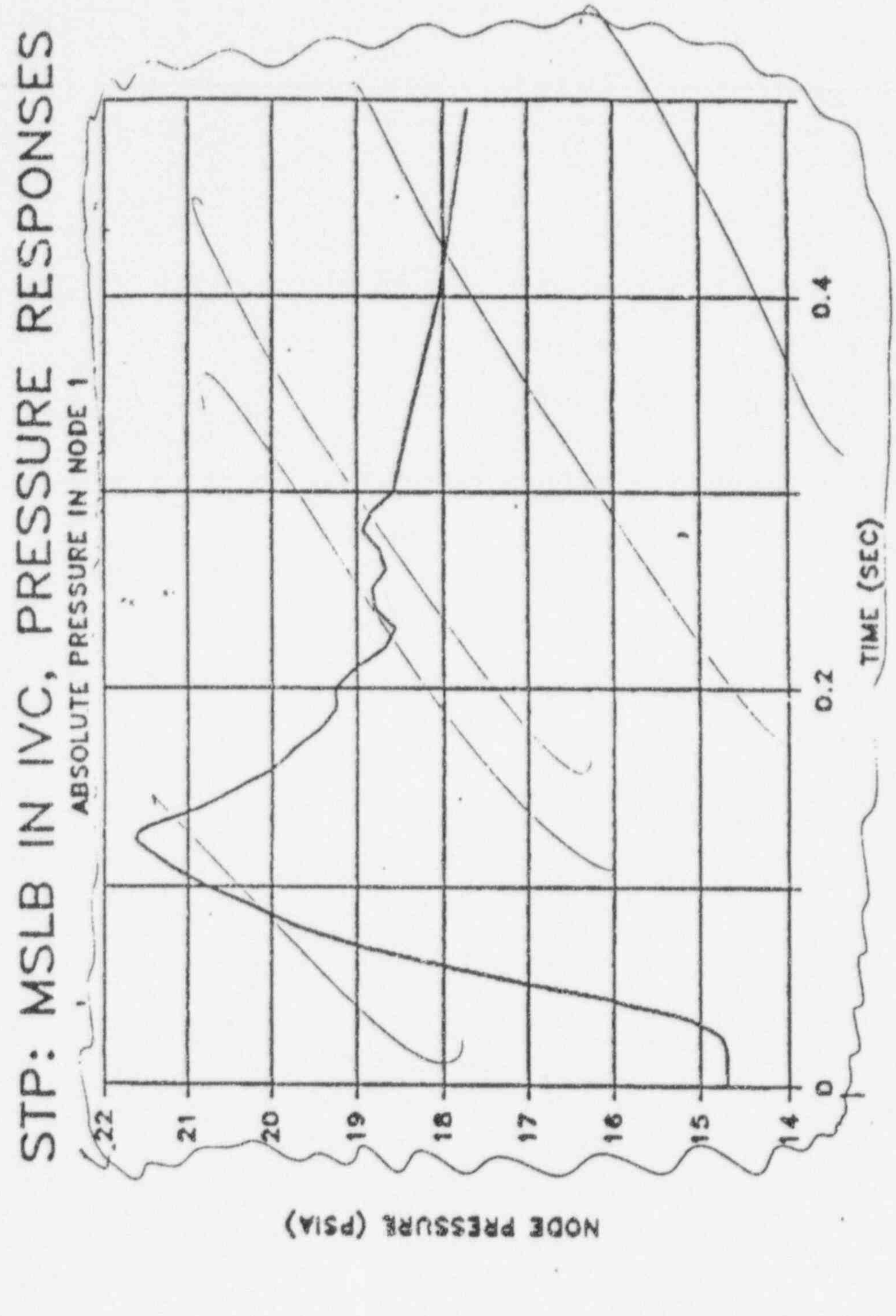
NODE AND JUNCTION DIAGRAM
OF THE IVC
FOR COPDA

Figure 3.6.A-1 A Revision 0



NODE AND JUNCTION DIAGRAM
OF THE IVC
FOR GOTHIC

FIGURE 3.6.A-1B REVISION C



Replace with attached Figure 1

SOUTH TEXAS PROJECT
UNITS 1 & 2

PRESSURES IN IVC DUE TO MSLB
NODE 1 (BREAK NODE ~~7~~ 6: CASE ~~2~~ 1)
(Sheet 1 of 9)

Figure 3.6.A-3 | Revision 0

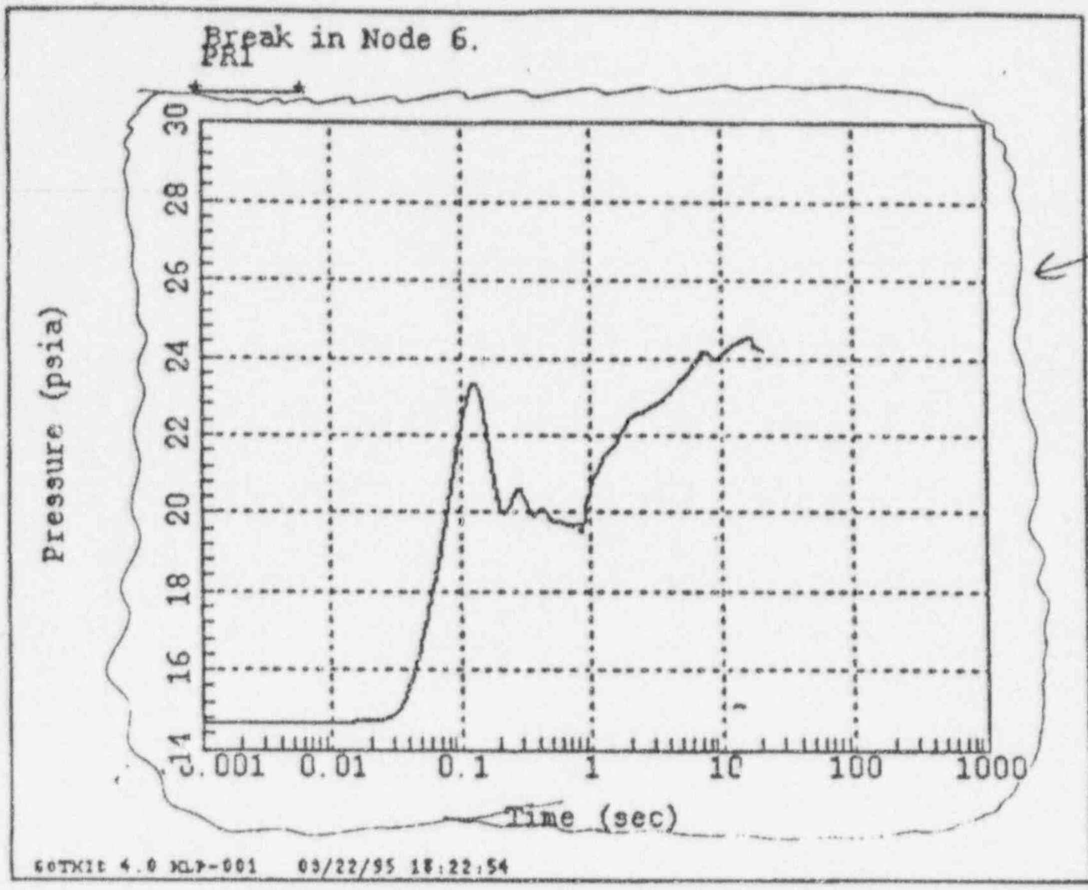
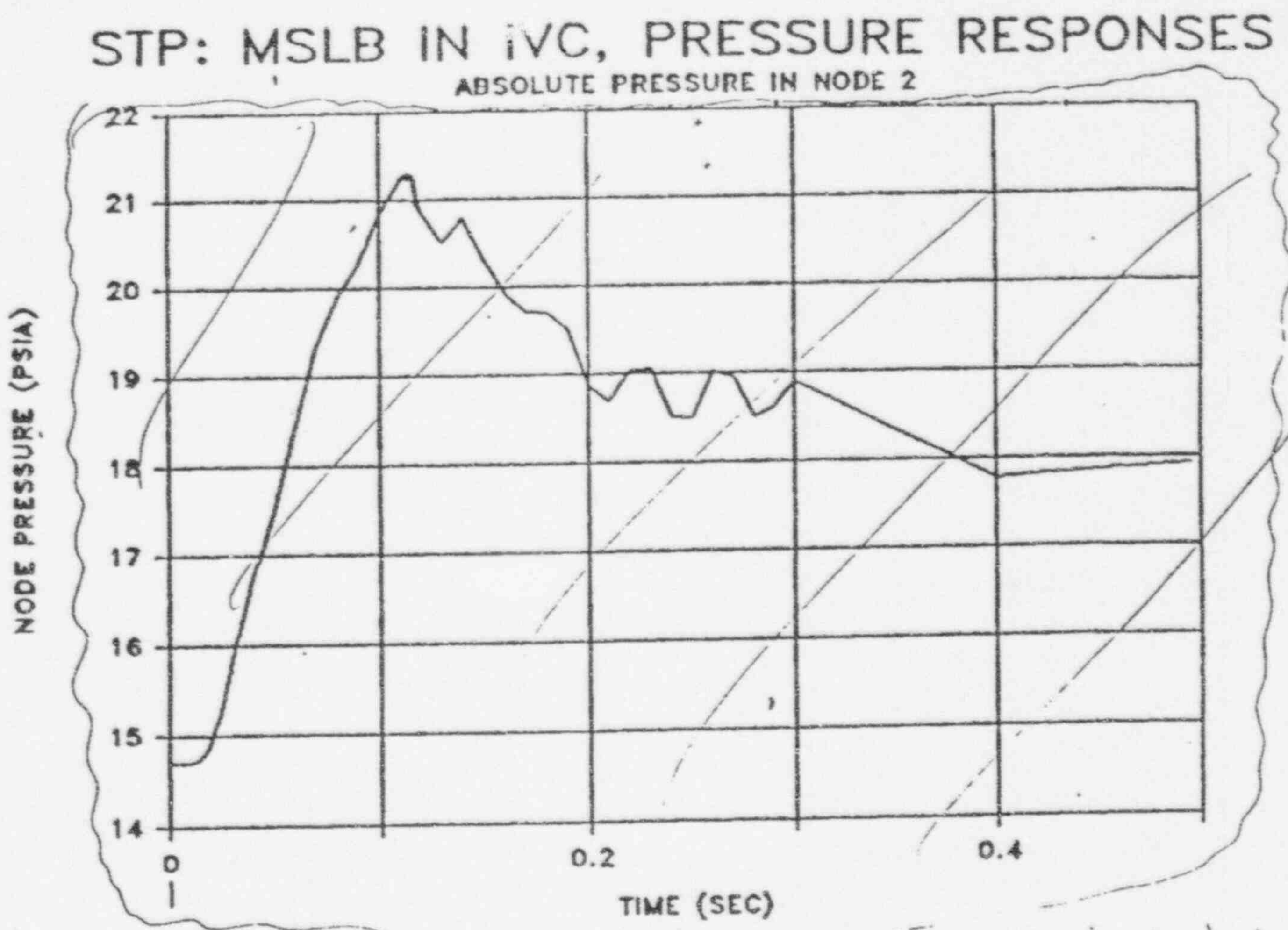

Break in Node 6.
PRI
Pressure (psia)
14
16
18
20
22
24
26
28
30
.001
0.01
0.1
1
10
100
1000
Time (sec)
GOTHIC 4.0 KLP-001 03/22/95 18:22:54

Figure 1



Replace with attached Figure2

SOUTH TEXAS PROJECT
UNITS 1 & 2

PRESSURES IN IVC DUE TO MSLB
NODE 2 (BREAK NODE ~~1~~: CASE ~~2~~) 1
(Sheet 2 of 9)

Figure 3.6.A-3

Revision 0

Base IVCSPP Model with Break in Node 6.
Wed Mar 22 19:18:06 1995
GOTHIC Version 4.0 HLP-001 - October 1994

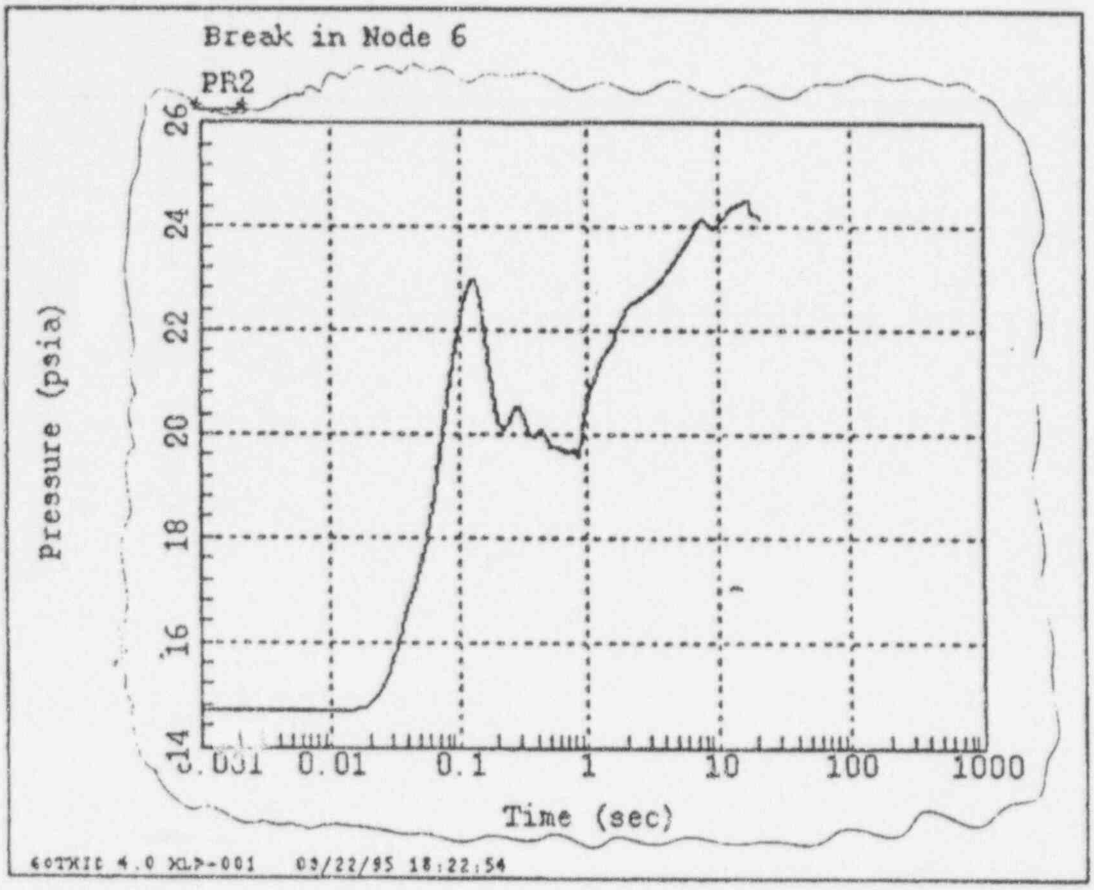
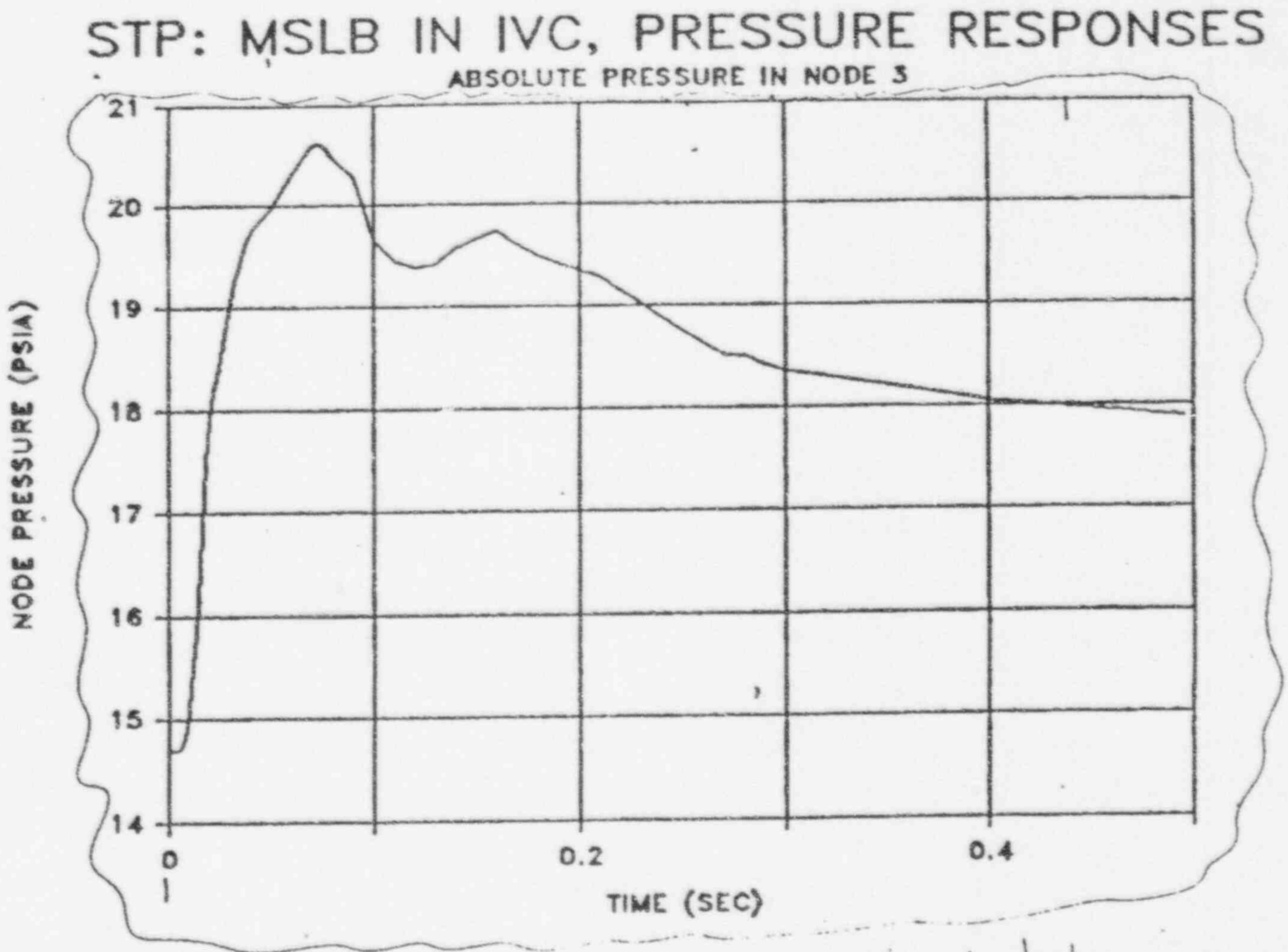


Figure 7



Replace with attached Figure 3

SOUTH TEXAS PROJECT
UNITS 1 & 2

PRESSURES IN IVC DUE TO MSLB
NODE 3 (BREAK NODE 6: CASE 1)
(Sheet 3 of 9)

Figure 3.6.A-3

Revision 0

Base IVCSPP Model with Break in Node 6.
Wed Mar 22 19:17:48 1995
GOTHIC Version 4.0 HLP-001 - October 1994

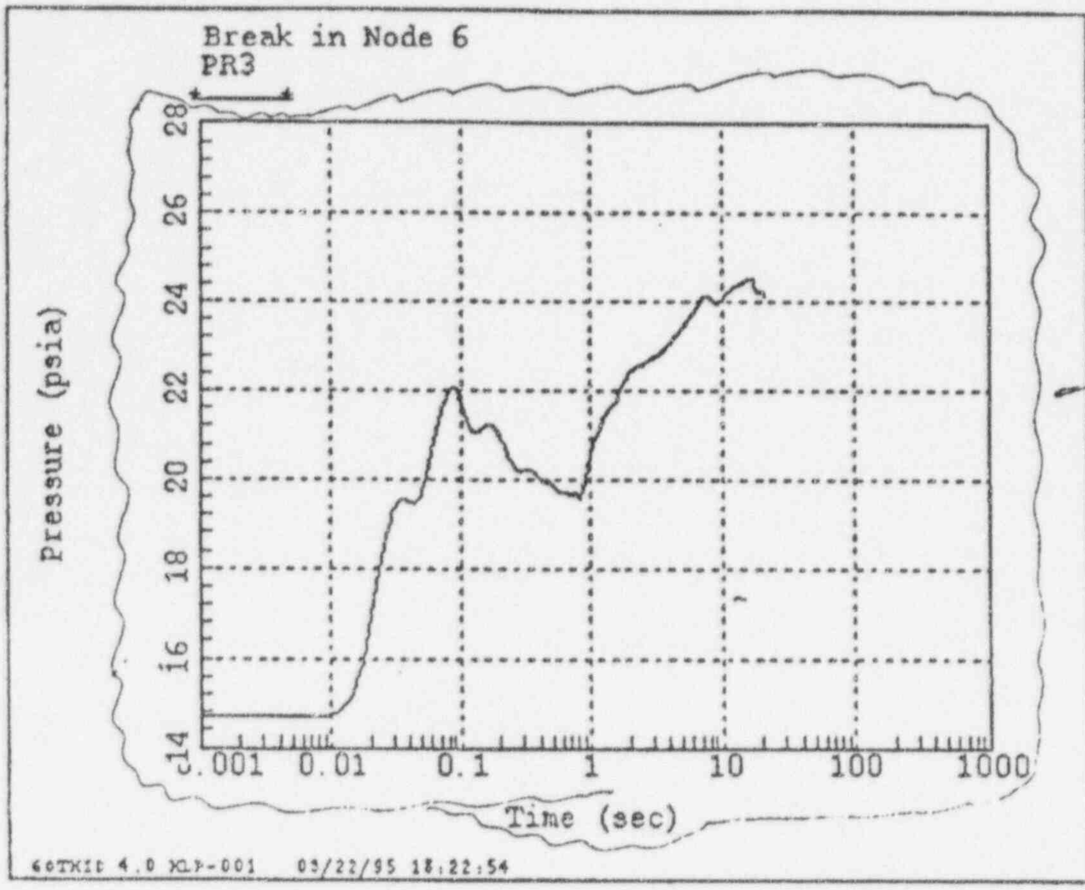
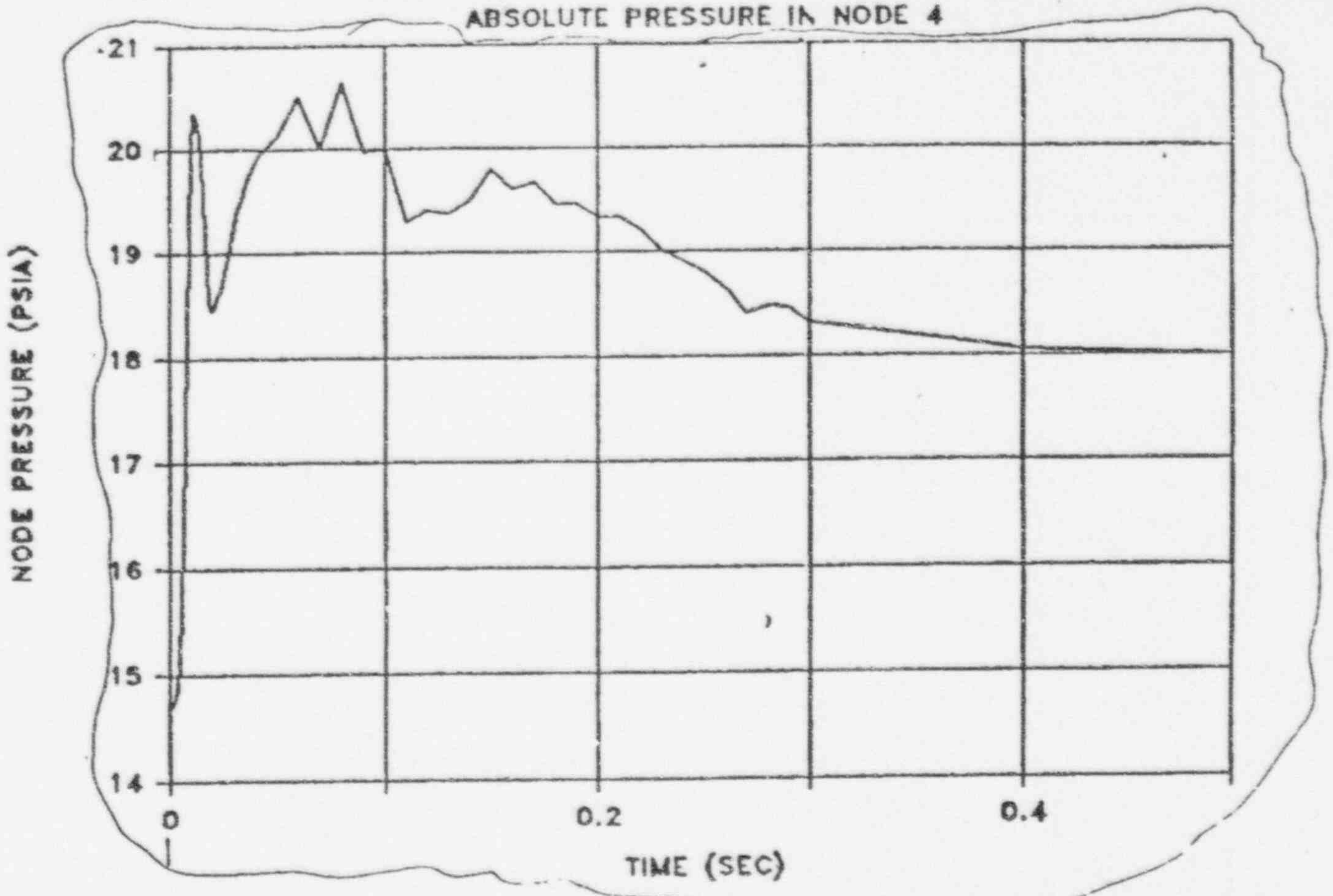


Figure 3

STP: MSLB IN IVC, PRESSURE RESPONSES



Replace with attached Figure 4

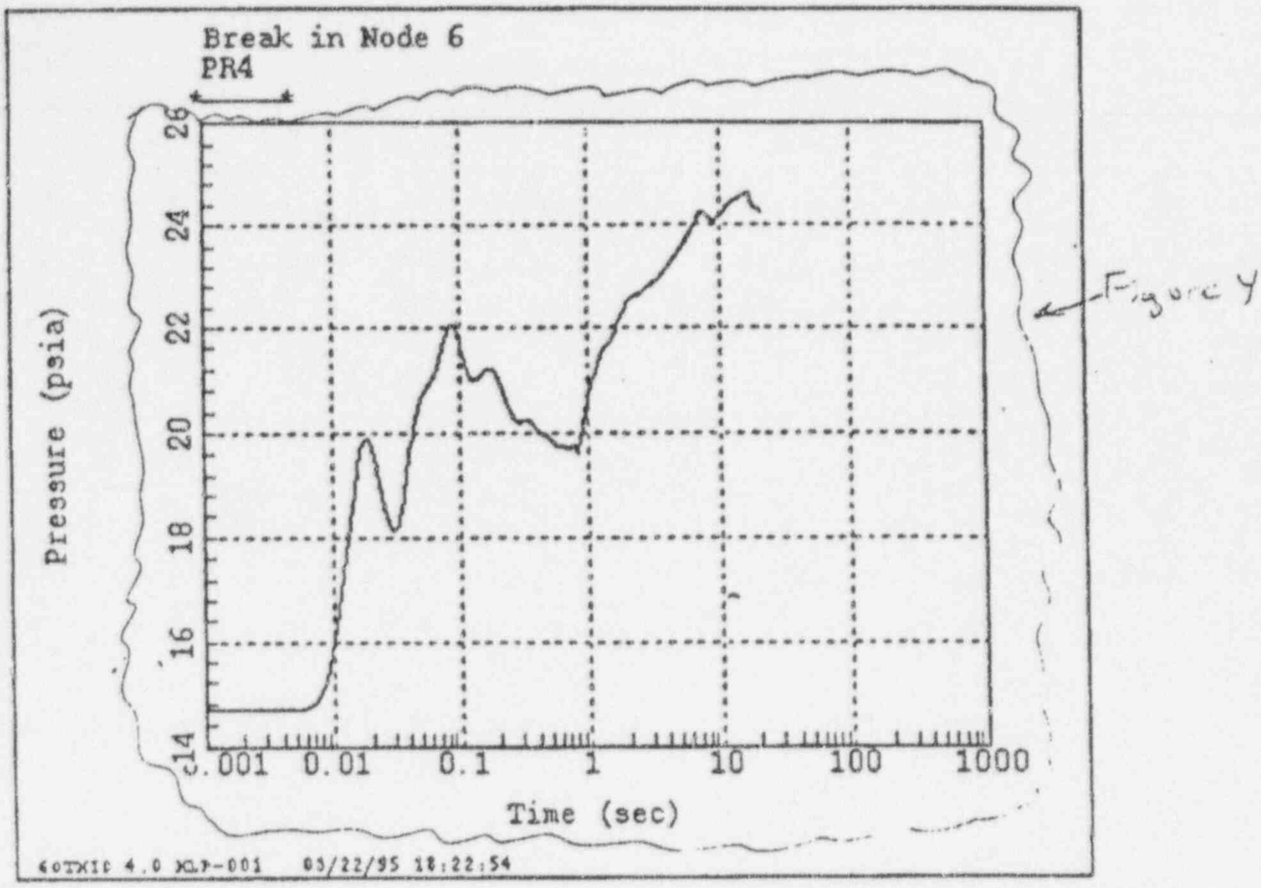
SOUTH TEXAS PROJECT
UNITS 1 & 2

PRESSURES IN IVC DUE TO MSLB
NODE 4 (BREAK NODE 6: CASE 1)
(Sheet 4 of 9)

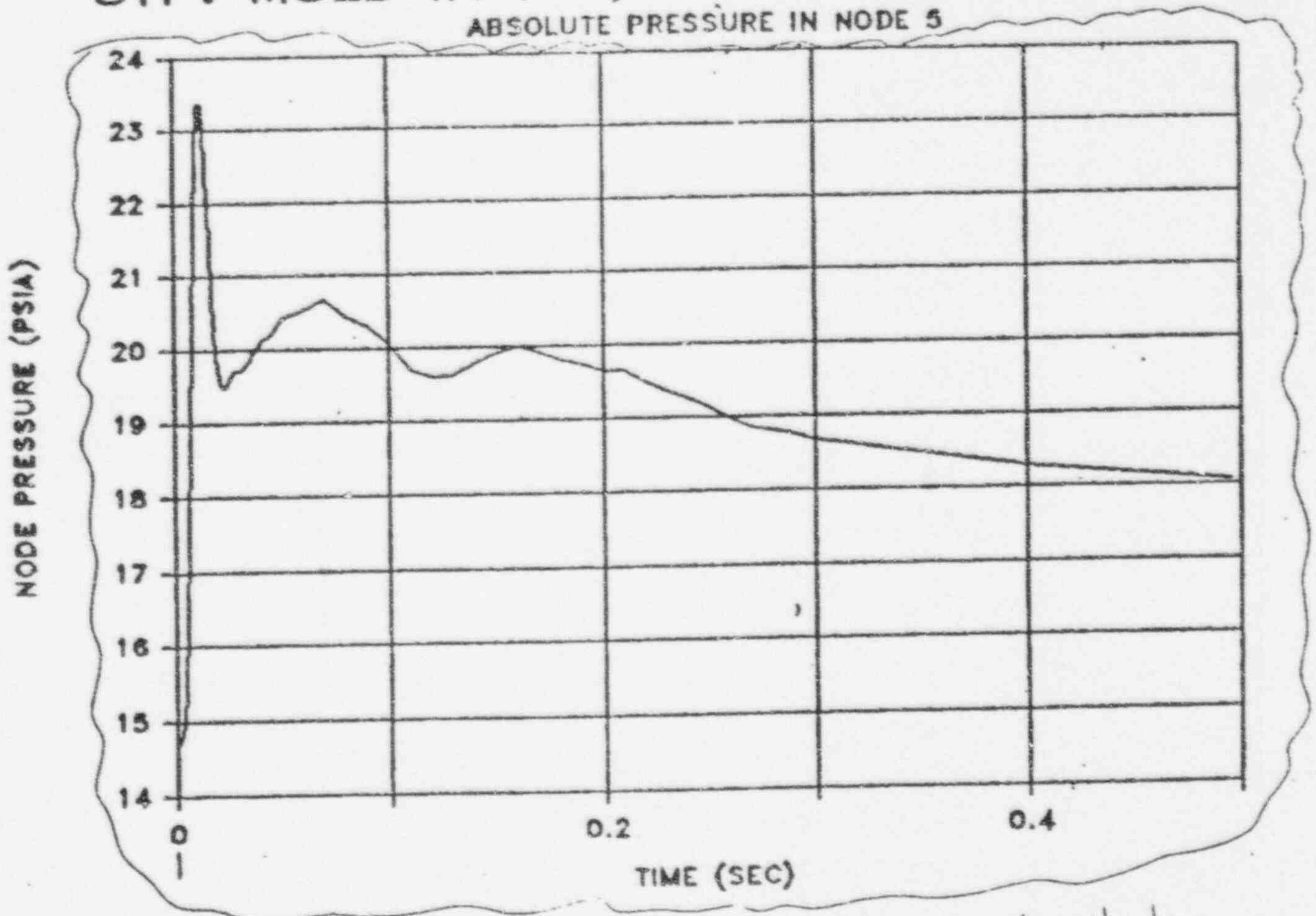
Figure 3.6.A-3

Revision 0

Base IVCSPP Model with Break in Node 6.
Wed Mar 22 19:17:55 1995
GOTHIC Version 4.0 HLP-001 - October 1994



STP: MSLB IN IVC, PRESSURE RESPONSES



Replace with attached Figure 5

SOUTH TEXAS PROJECT
UNITS 1 & 2

PRESSURES IN IVC DUE TO MSLB
NODE 5 (BREAK NODE 7: CASE 2)
(Sheet 5 of 9)

Figure 3.6.A-3

Revision 0

Base IVCSPP Model with Break in Node 7.
Wed Mar 22 19:37:57 1995
GOTHIC Version 4.0 HLP-001 - October 1994

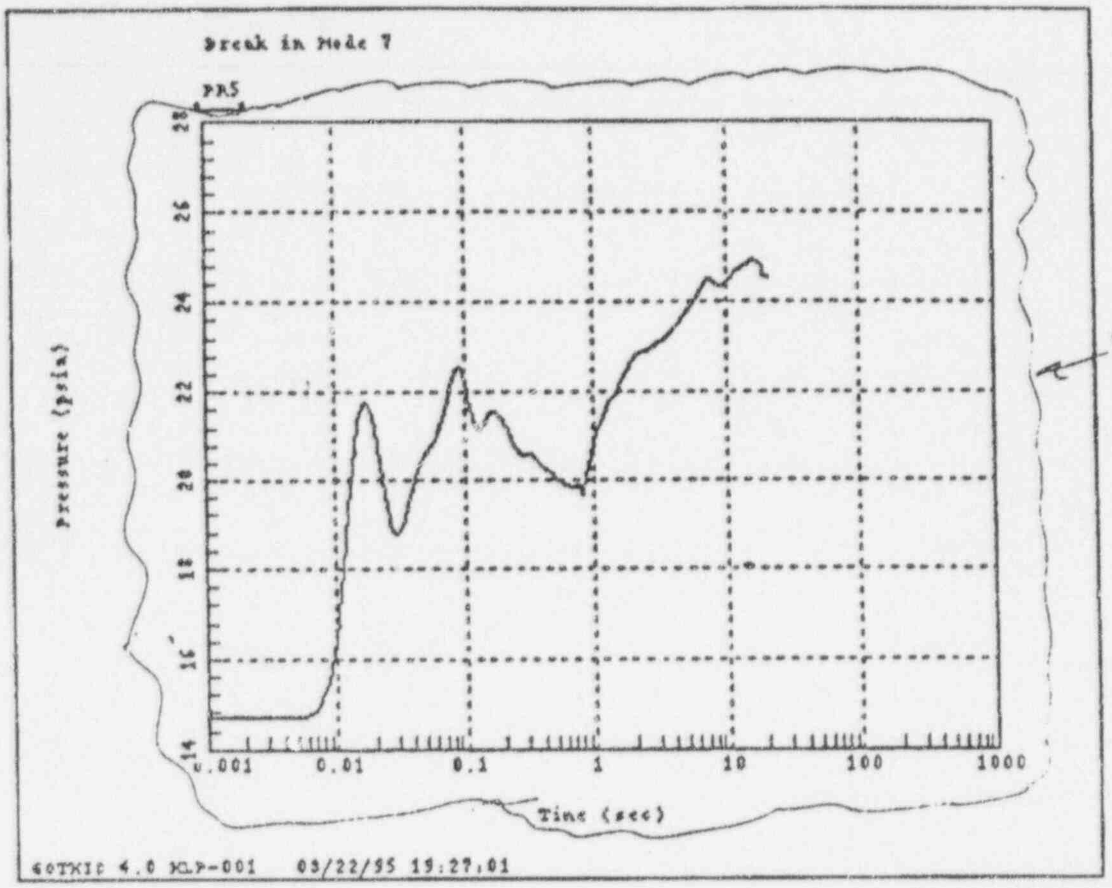
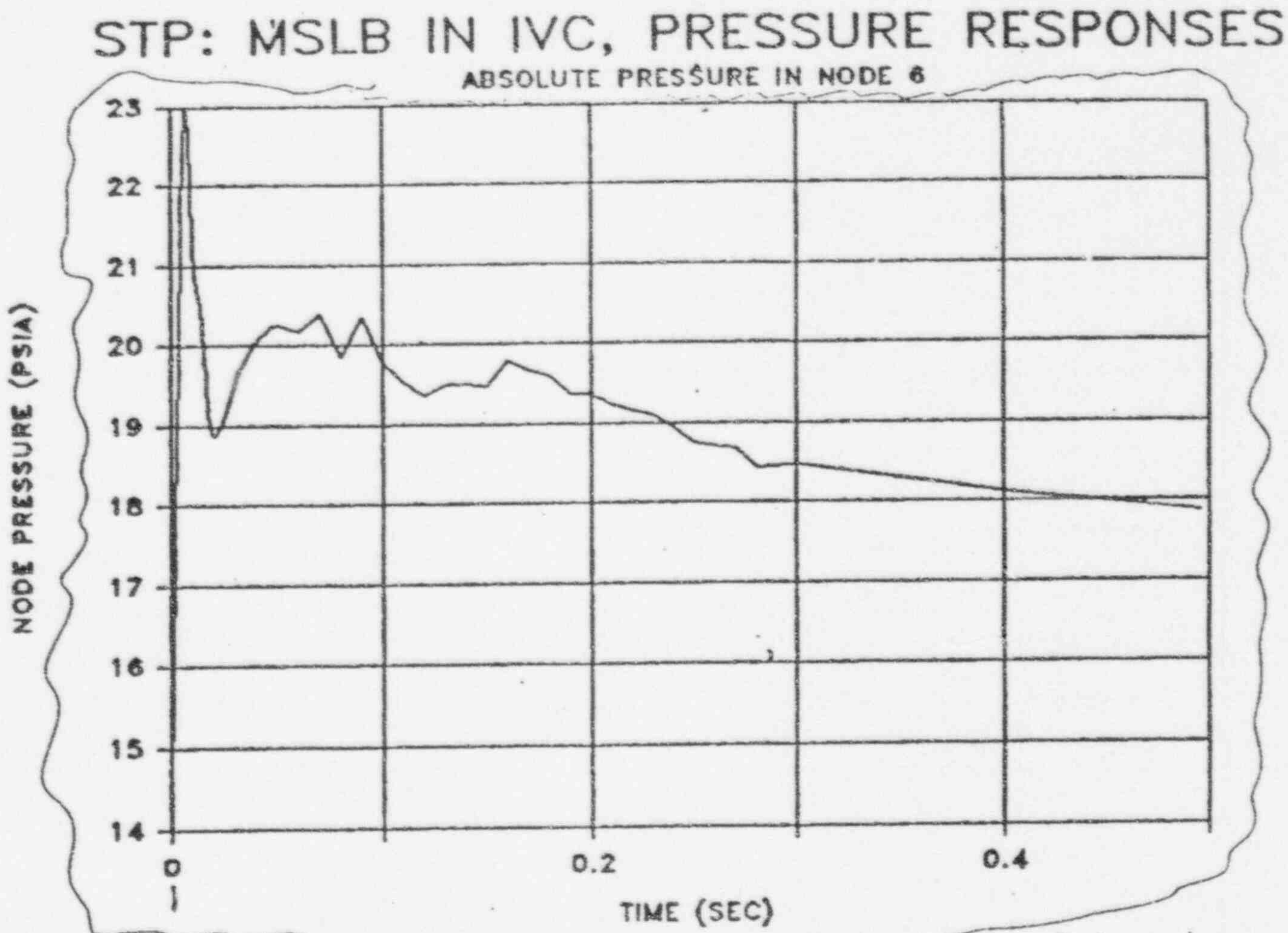


Figure 5



Replace with attached Figure

SOUTH TEXAS PROJECT
UNITS 1 & 2

PRESSURES IN IVC DUE TO MSLB
NODE 6 (BREAK NODE 6: CASE 1)
(Sheet 6 of 9)

Figure 3.6.A-3

Revision 0

Base IVCSPP Model with Break in Node 6.
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GOTHIC Version 4.0 HLP-001 - October 1994

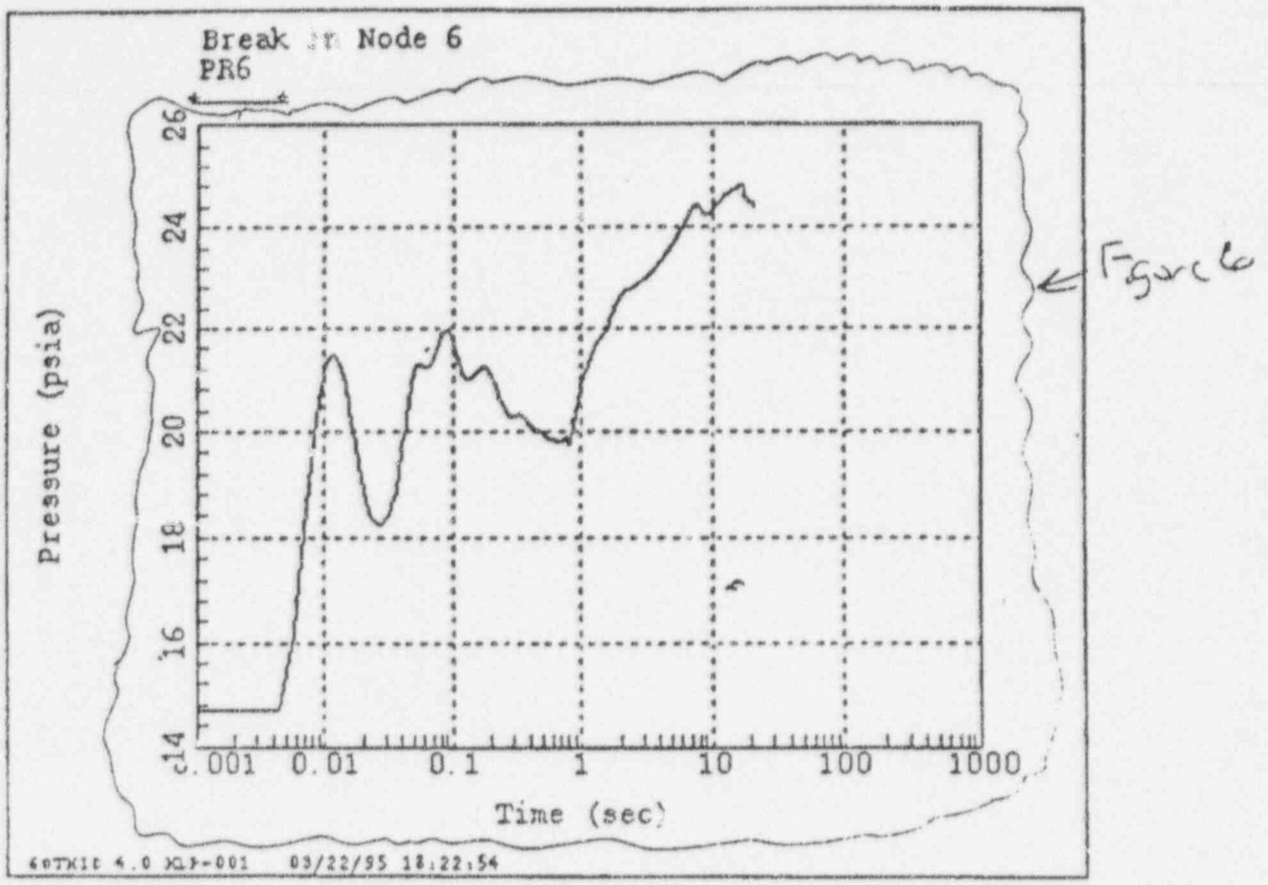
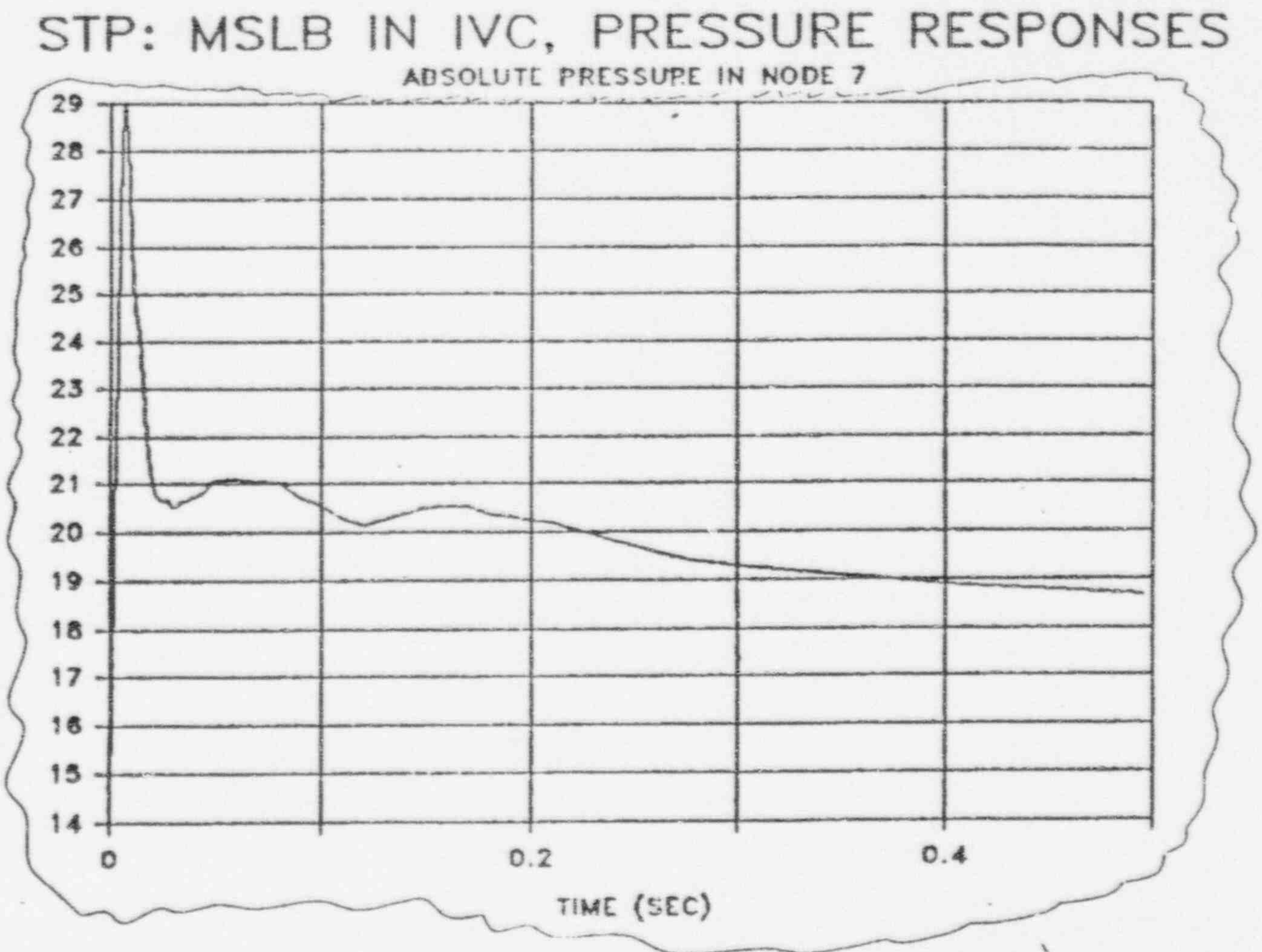


FIG. 4.2-1



Replace with Attached Figure 7

SOUTH TEXAS PROJECT
UNITS 1 & 2

PRESSURES IN IVC DUE TO MSLB NODE 7 (BREAK NODE 7: CASE 2)
(Sheet 7 of 9)

Figure 3.6.A-3

Revision 0

Base IVCSPP Model with Break in Node 7.
Wed Mar 22 19:38:00 1995
GOTHIC Version 4.0 HLP-001 - October 1994

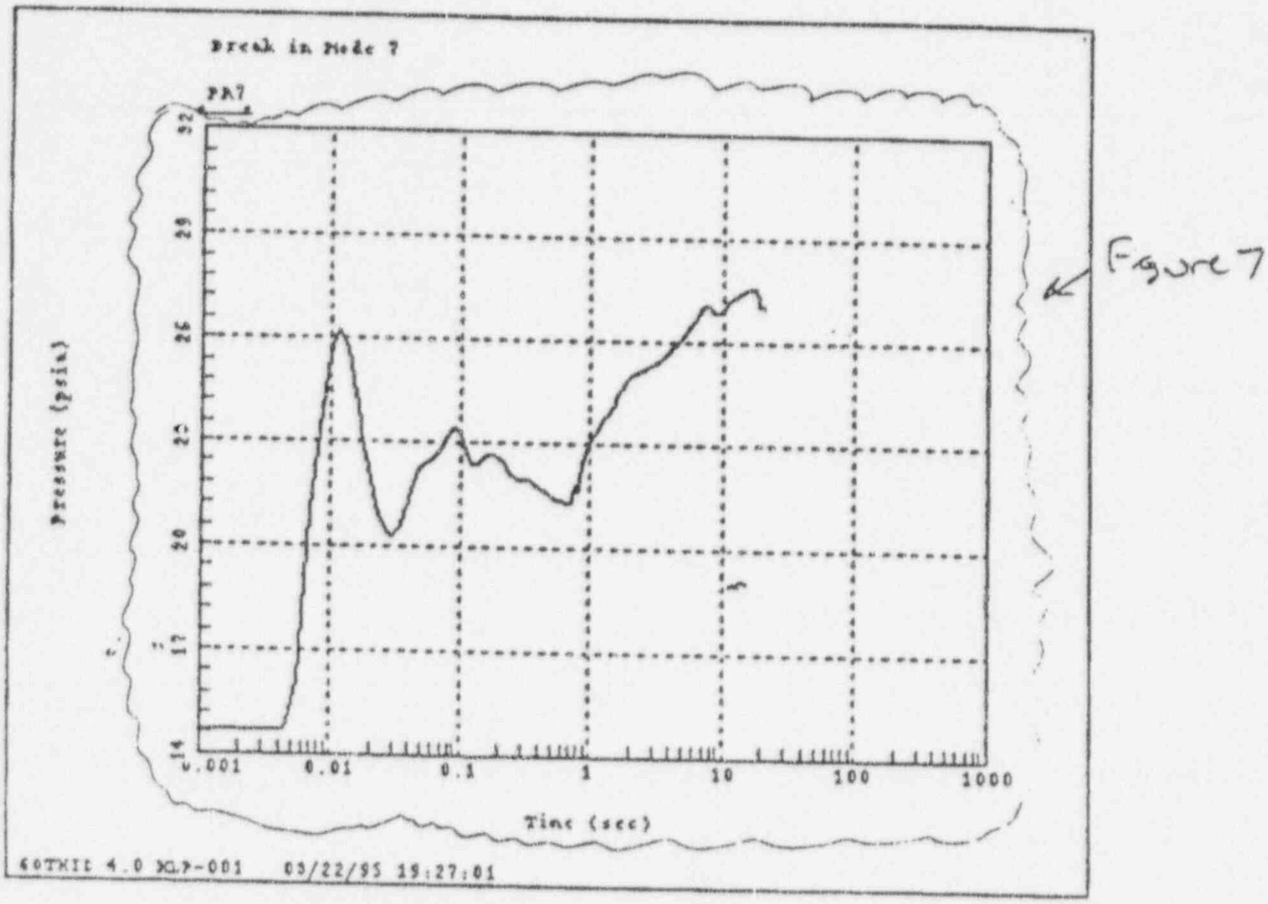
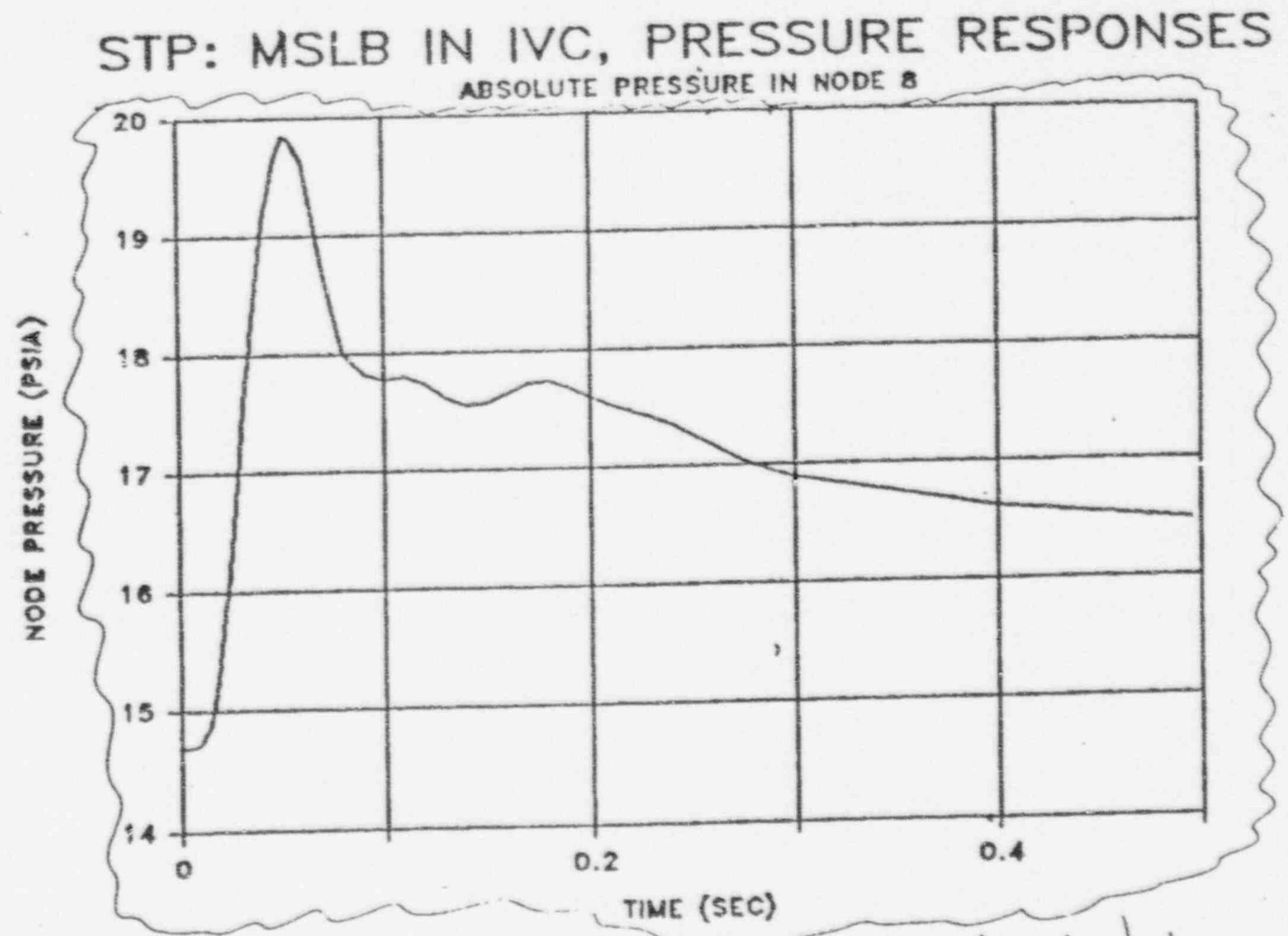


FIG. 4.2-3



↑ Replace with attached Figure B

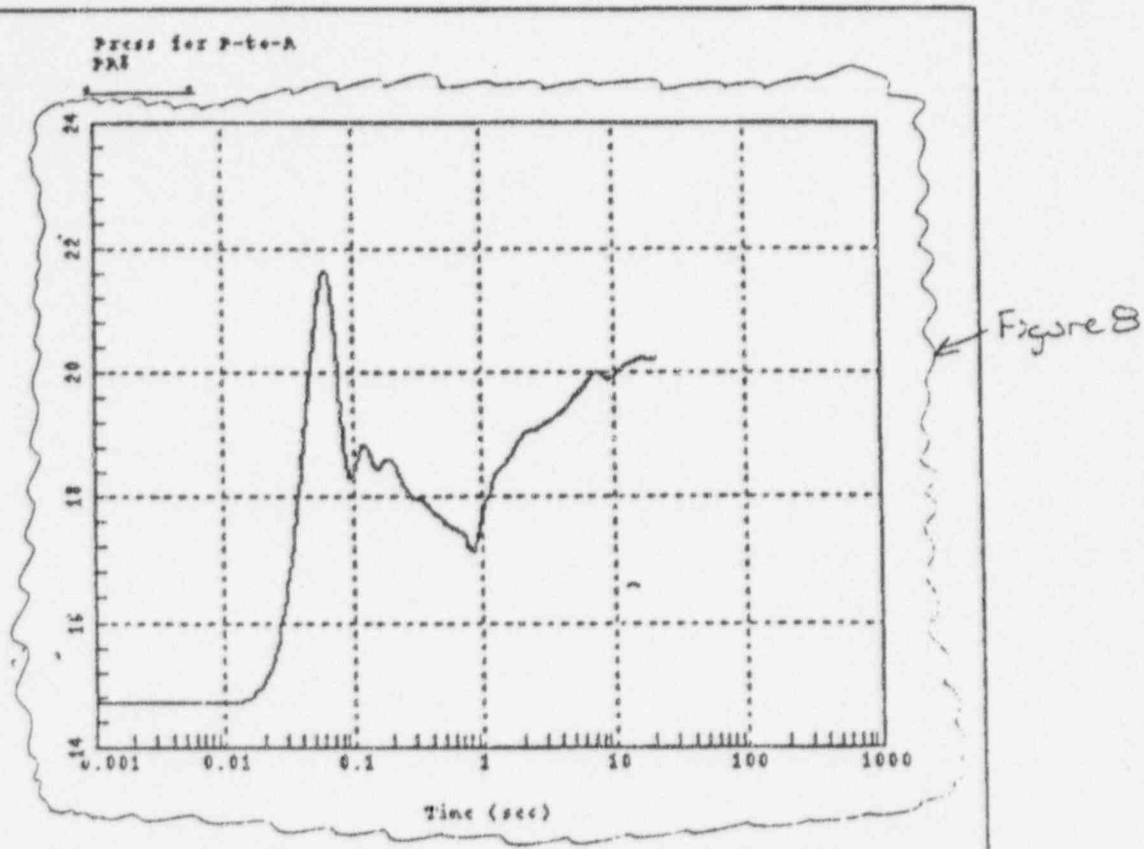
SOUTH TEXAS PROJECT
UNITS 1 & 2

PRESSURES IN IVC DUE TO MSLB
NODE 8 (BREAK NODE 7: CASE 2)
(Sheet 8 of 9)

Figure 3.6.A-3

Revision 0

Base IVCSPP Model with Break in Node 7.
Wed Mar 22 19:37:52 1995
GOTHIC Version 4.0 HLP-001 - October 1994



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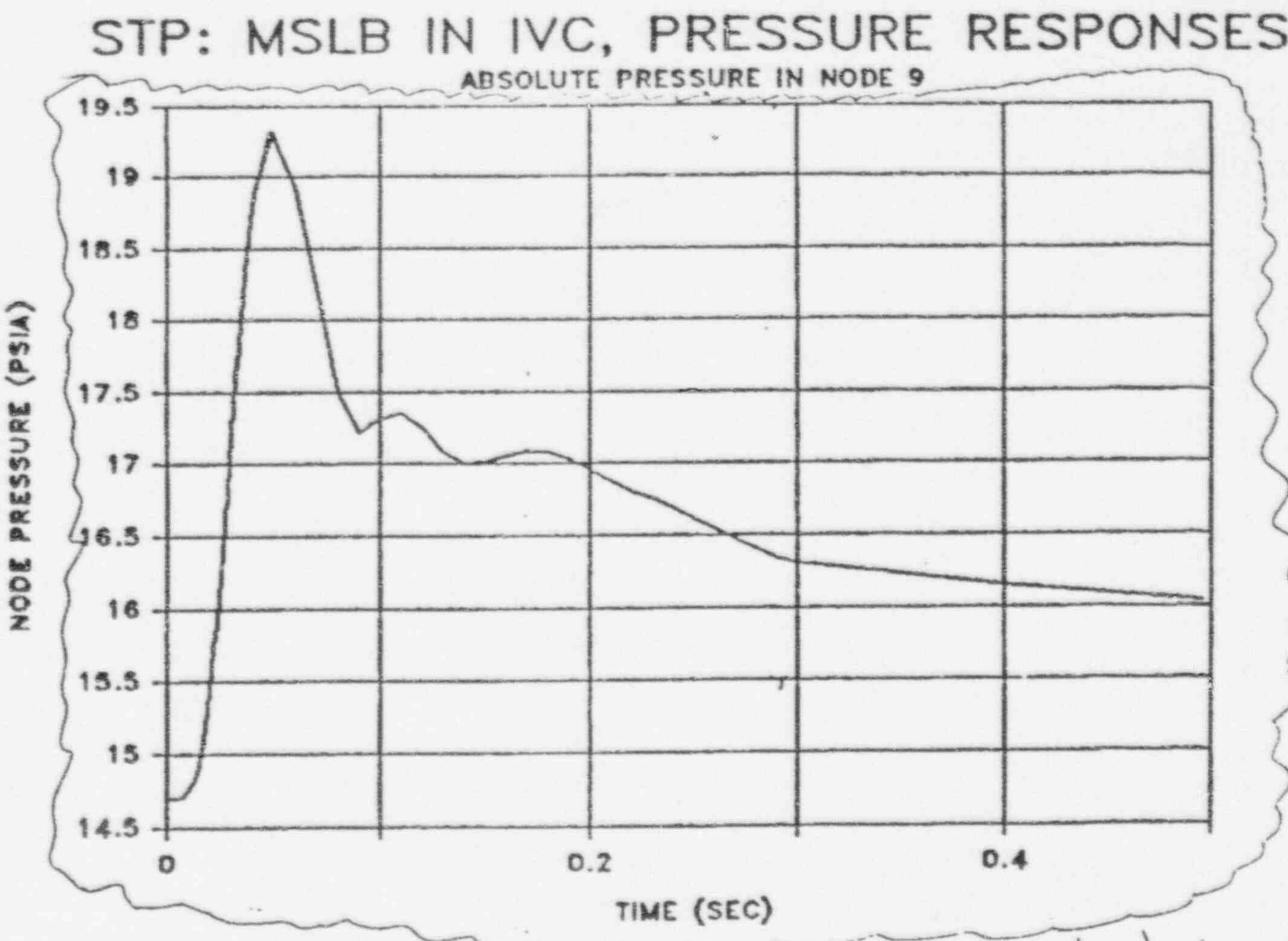


Replace with the attached Figures

SOUTH TEXAS PROJECT
UNITS 1 & 2

PRESSURES IN IVC DUE TO MSLB
NODE 9 (BREAK NODE 6: CASE 1)
(Sheet 9 of 9)

Figure 3.6.A-3

Revision 0

Base IVCSPP Model with Break in Node 6.
Wed Mar 22 19:17:59 1995
GOTHIC Version 4.0 HLP-001 - October 1994

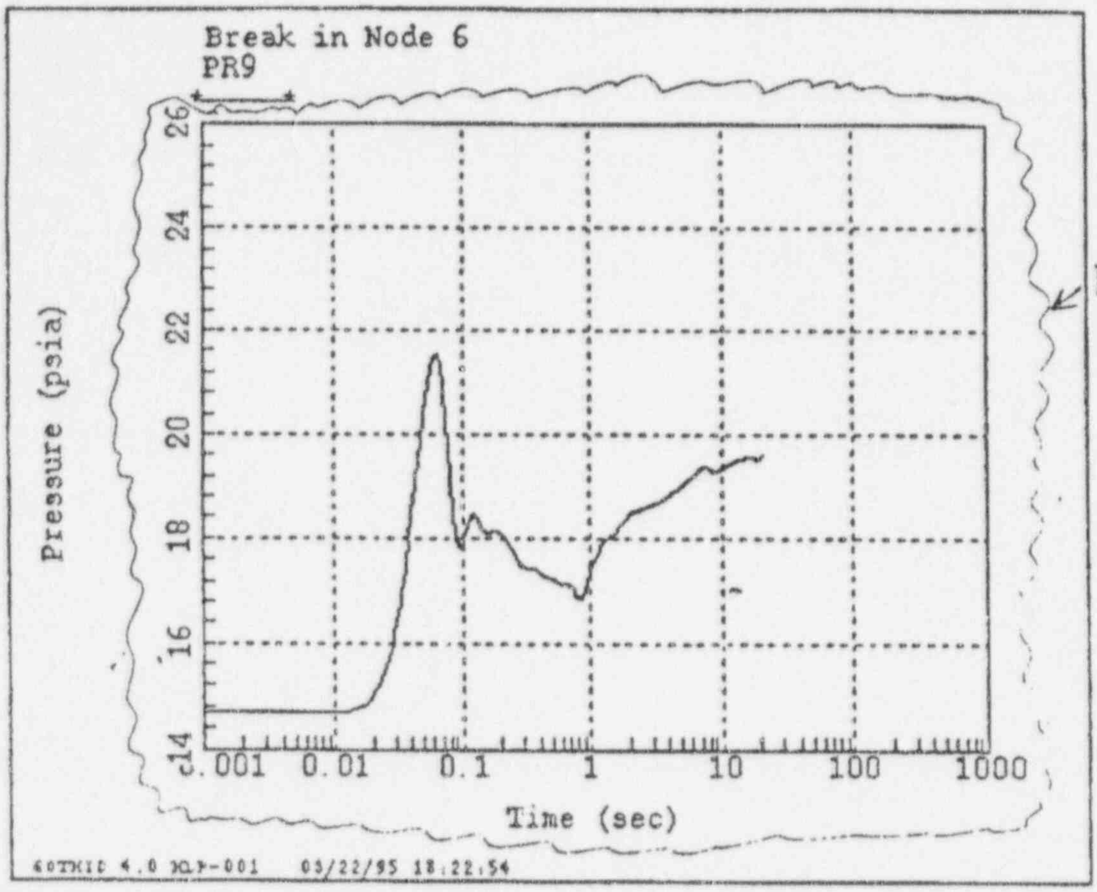


Figure 9

blowout panels are used, thus the flow area is assumed to be constant with respect to time.

6.2.1.2.3 Design Evaluation:

(with the exception of the Mainsteam and Feedwater line Subcompartments,

6.2.1.2.3.1 General - The subcompartment pressure transients were determined using the COPDA computer code (Ref. 6.2.1.2-1). The COPDA code employs a finite difference technique to solve the time dependent equations for the conservation of mass, energy and momentum. This code and the assumptions inherent to it are fully explained in Reference 6.2.1.2-2. Loss coefficients utilized were based on the formulations of References 6.2.1.2-3 and 6.2.1.2-4.

Insert 1

Nodalization of each subcompartment was based on the physical arrangement of the interconnected subcompartment and the structure, equipment, piping, ventilation ducting, floor grating, and other physical obstructions to flow. By appropriate selection of node boundaries based on the physical arrangement, pressure differences within a node are minimized while pressure differences between nodes are maximized.

The LOCA blowdown model used to calculate the short-term mass and energy release rates for all primary system ruptures, including the surge line break and the pressurizer spray line break, is fully described in Reference 6.2.1.2-5. The mass and energy release data are presented in Table 6.2.1.2-1.

INSERT 2

The RELAP 5 code (Ref. 6.2.1.2-6) was used to calculate the short-term blowdown of the main steam line and main feedwater line. The mass and energy release rates for these two lines are provided in Table 6.2.1.2-1. Letdown line break blowdown was calculated using the American National Standards Institute (ANSI) methodology of Reference 6.2.1.2-7.

The ANSI methodology for subcooled blowdown from a pipe break results in a decompression wave propagating through the system at sonic velocity with the pressure behind the wave corresponding to saturation pressure of the liquid. Because of the very low compressibility of subcooled water, subcooled blowdown cannot be sustained for more than a few milliseconds and the total mass release under subcooled blowdown conditions is quite small. Following this extremely short-term initial phase, the pressure will correspond to saturation pressure. The blowdown for saturated and subcooled water conditions is determined using Henry-Fauske and Moody relationships and is given in Table 6.2.1.2-1.

6.2.1.2.3.2 Reactor Cavity - No pipe breaks are postulated in the reactor cavity and inspection toroid.

6.2.1.2.3.3 Steam Generator Subcompartment - SG subcompartment design pressure is determined by breaks in the pressurizer surge line and SI accumulator injection lines. The blowdown for these breaks is presented on Table 6.2.1.2-1. The noding of the SG compartments is shown on Figure 6.2.1.2-3. The node and junction diagram is shown on Figure 6.2.1.2-11. The flow parameters were evaluated to account for all obstructions such as cable tray supports and various small-sized piping. The principal obstructions within the SG loop compartments are the SG and reactor coolant pumps. The NRC-approved COPDA computer program (Ref. 6.2.1.2-2) was used to perform the subcompartment analyses. The modeling for the COPDA program requires that

INSERT 1

The main steam and feedwater subcompartment pressure transients were determined using the GOTHIC 4.0 computer code. A description of this computer code is presented in Section 3.6.A.6.

INSERT 2

The RETRAN-03 computer code was used to calculate the short term mass and energy release of the main steam line. The RELAP 05 computer code (Ref 6.2.1.2-6) was used to calculate the short term mass and energy release of the main feedwater line. The mass and energy release rates for the main feedwater line is presented in Table 6.2.1.2-1. Letdown line break mass and energy release was calculated using the American National Standards (ANSI) methodology of Reference 6.2.1.2-7.

6.2.1.2.3.6 Main Steam and Feedwater Line Subcompartments - The main steam and feedwater line subcompartments are shown on Figure 6.2.1.2-6, with a node and junction diagram given on Figure 6.2.1.2-14. A double-ended main steam line rupture (8.1 ft²) was assumed to occur in either Node 1 or 3. The calculated peak pressure occurred in Node 3.

A double-ended rupture (2.837 ft²) of the main feedwater line was assumed to occur in either Node 5 or 7. The calculated peak pressure occurred in Node 7.

Insert 3 →

Node and junction parameters utilized in the analyses are given in Tables 6.2.1.2-13 and 6.2.1.2-14. Plots of calculated pressures are given on Figures 6.2.1.2-25 and 6.2.1.2-26, while calculated and design values are compared in Table 6.2.1.2-13. Mass and energy release rates are provided in Table 6.2.1.2-1. ~~The mass and energy release rates are calculated using RELAP-5 analysis.~~

Insert 4.

6.2.1.2.3.7 Regenerative Heat Exchanger Subcompartment - A double-ended rupture of the CVCS letdown line is the limiting break in the regenerative heat exchanger subcompartment. A node and junction diagram is given on Figure 6.2.1.2-15. The nodal model initial conditions, control volumes, vent areas and corresponding flow coefficients and inertial terms are given in Tables 6.2.1.2-15 and 6.2.1.2-16. The calculated subcompartment pressure response is shown on Figure 6.2.1.2-27. Calculated and design pressures are compared in Table 6.2.1.2-15. The blowdown rate for the CVCS letdown line break is calculated using ANSI 58.2, Appendix E2, methodology (Ref. 6.2.1.2-7) and applying that to a one-dimensional Henry-Fauske model for saturated liquid. Mass and energy release rates are shown in Table 6.2.1.2-1 (Refer to Section 6.2.1.2.3.1 for more details). Plant operation is assumed to be in the heat-up mode. The break is assumed to occur at the inlet to the regenerative heat exchanger. The break area is 0.0884 ft² for each end of the double-ended break (0.1768 ft² total area). There are no significant restrictions to forward flow, but the reverse flow is restricted by the CVCS letdown orifices (0.00166 ft²) located immediately downstream of the regenerative heat exchanger. In addition, the reservoir of reverse flow is limited since high energy fluid conditions extend only to the letdown heat exchanger.

6.2.1.2.3.8 Radioactive Pipe Chase Subcompartment - A double-ended rupture of the CVCS letdown line is the limiting break in the radioactive pipe chase subcompartment. A node and junction diagram is illustrated on Figure 6.2.1.2-16. The flow model initial conditions, control volumes, inter-compartment flow paths, and corresponding flow coefficients and inertial terms are listed in Tables 6.2.1.2-17 and 6.2.1.2-18. The calculated subcompartment pressure response is shown on Figure 6.2.1.2-28. The calculated and design pressures are compared in Table 6.2.1.2-17. The blowdown rate for the CVCS letdown line break is calculated using ANSI 58.2, Appendix E2 methodology and applying that to a one-dimensional Henry-Fauske model for saturated liquid. Mass and energy release rates are given in Table 6.2.1.2-1 (Refer to Section 6.2.1.2.3.1 for more details). Plant operation is assumed to be in the heatup mode. The break is assumed to occur at the Containment penetration. The break area is 0.0884 ft² for each end of double-ended break (0.1768 ft² total area). A significant restriction to forward flow is the CVCS letdown orifices (0.00166 ft²) located immediately downstream of the regenerative heat exchanger. For reverse flow, the letdown heat exchanger reduces the line temperature to 115°F and a pressure reducing valve, immediately downstream of

INSERT 3

Results of the analysis show that the main steam line break bounds the main feedwater line break case.

INSERT 4

The mass and energy releases were calculated using the RETRAN-03 computer code as discussed in Section 6.2.1.4.8.

2. Failure of Main Feedwater Pump Trip

No credit is taken for feedwater pump trip and coastdown in calculating main feedwater addition prior to feedwater line isolation. Therefore, this failure has no effect on the results presented.

3. Failure of a Main Steam Line Isolation Valve

Failure of an MSIV is assumed to increase the volume of steam piping which empties into the Containment by 5,570 ft^3. This case is included in the analysis.

The effects of this failure on calculated Containment pressures and temperatures were compared with the effects of the failure of one Containment spray train. With respect to the maximum Containment pressure, calculations showed that the adverse effects of a main steam line isolation valve failure were considerably less than that of one Containment spray train failure. With respect to the maximum Containment temperature, no significant difference was found between the two failures.

4. Failure of One Containment Heat Removal Train

The worst single failure following a steam line break is the failure of one of the three redundant CHRS trains. The spray actuation is assumed to occur 69.1 seconds following the time at which the Containment pressure reaches ?2.0 psig. The fan cooler actuation is assumed to occur 30.0 seconds after the Containment pressure reaches its setpoint.

6.2.1.4.6 This section is not used.

Insert 5

~~6.2.1.4.7 Short Term Steam Line and Feedwater Line Breaks: The RELAP 5/MOD1 (Ref. 6.2.1.2-6) computer program was used for the secondary system pipe break analysis. The initial conditions for the main steam system are taken to be at full power (100 percent) operating conditions (1,100 psia and 556.6°F). Some of the major assumptions made in the analysis, all of them conservatively maximizing the blowdown, are the following:~~

1. ~~The postulated high energy line double-ended rupture is assumed to reach its maximum open area (with each pipe end discharging through a break area equal to the internal cross-sectional area of the pipe) within one millisecond of break initiation.~~

2. ~~During the course of the transient, the SG pressure and temperature are assumed to remain constant at the initial conditions.~~

3. ~~All four MSIVs remain wide open for the duration of the transient. Since the simulation is carried out to one second only, the transient will have ended before the MSIVs receive the signal to close.~~

4. ~~The sink volume is maintained at a constant pressure and temperature of 14.7 psia and 120°F, respectively, with the noncondensible air quality pegged at 0.98.~~

5. A conservative assumption that the turbine generator will remain at 1,100 psia and 556.6°F is made to maximize the reverse flow blowdown.

6. Smooth commercial steel pipe, with a relative roughness of 0.00015 ft, is assumed for all piping.

7. A throat with an area of 1.3882 ft^2 is assumed to limit the blowdown from the SG side.

For the short-term feedwater line break analysis, the assumptions included are:

1. Pipes are assumed to be commercial steel with a roughness of 0.00015 ft

2. An inlet area of 0.22 ft^2 for feedwater inlet at the SG nozzle is used in the analysis

3. A one millisecond break opening time is used in the study

4. A maximum flow area of 90 percent of the pipe area was assumed and 100 percent of the pipe area was used for the feedwater isolation valve (FW)

5. The piping and other elements of the RELAP model are assumed to be at a pressure of 1,135.0 psia and a temperature of 440°F

6. The break junctions are releasing mass into an environment that is maintained at a constant temperature and pressure of 120°F and 14.7 psia

7. Feedwater pump characteristics are not included in this analysis

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies of Emergency Core Cooling System. The Containment backpressure used for the limiting case C_D=0.6 (max. SI), double-ended cold leg guillotine break for the ECCS analysis presented in Section 15.6.5 is presented on Figure 6.2.1.5-1. Containment backpressure is calculated using the methods and assumptions described in Appendix A of Reference 6.2.1.5-1. Input parameters used in the analysis, including Containment initial conditions, net free Containment volume, passive heat sink materials, thicknesses, and surface areas, and starting time and number of Containment cooling systems, are described below.

6.2.1.5.1 Mass and Energy Release Data: The mass and energy releases to the Containment during the blowdown portion of the limiting break transient are shown in Table 6.2.1.5-1. The mass and energy flow rates during blowdown of the broken loop (safety injection is assumed to spill directly to Containment) are presented in Table 6.2.1.5-1A. Table 6.2.1.5-2 presents the mass and energy flow rates to Containment during the reflood portion of the transient.

The mathematical models that calculate mass and energy releases to the Containment are described in Section 15.6.5. Since the requirements of Appendix K of 10CFR50 are very specific as to the modeling of the RCS during blowdown, and since the models used are in conformance with Appendix K, no alterations to those models have been made with regard to the mass and energy releases. A break spectrum analysis is performed (see references in Section

INSERT 5

The RETRAN-03 (Ref. 6.2.1.2-8) computer code was used for the main steam line pipe break analysis. The RETRAN-03 computer code has been verified by HL&P for calculating short-term mass and energy release rate data following a postulated Main Steam Line Break for STPEGS. RETRAN-03 also meets the requirements of ANSI/ANS-56.10-1982 to perform this function.

RETRAN-03 is a best-estimate transient thermal-hydraulic code designed to analyze operational transients, small break loss-of-coolant accidents, anticipated transients without scram, natural circulation, long-term transients, and events involving limited nonequilibrium conditions in light water reactors. RETRAN-03 is a derivative of the RELAP4 code referred to in ANSI/ANS-56.10-1982. RETRAN-03 is the result of a program sponsored by the Electric Power Research Institute since 1975 to analyze thermal-hydraulic transients. Earlier versions of RETRAN-03 (e.g., RETRAN-02) have been reviewed by the NRC and received a Safety Evaluation Report.

Major assumptions of the main steam line break analysis are as follows:

1. The initial conditions for the main steam system are at zero power operating conditions plus instrument error (1266 psia and 574°F).

2. The postulated high energy line double-ended rupture is assumed to reach maximum opening area within one millisecond of break initiation. Each pipe end discharges through a break area equal to the internal cross-sectional area of the pipe.

3 During the transient, the SG pressure and temperature are assumed to remain constant at the initial conditions. This is conservative, because the actual SG pressure would decrease during this event.

4. The quality of the moisture carryover is conservatively assumed to be 4% and is assumed to continue until the mass of the affected steam generator (including AFW flow) is depleted. The 4% quality assumption is taken from Appendix E of ANSI 58.2-1980.

5. The analysis continues for greater than 20 seconds until the water mass in the affected steam generator (included AFW flow) is depleted. After the mass in the effected steam generator is depleted, the mass and energy release from this generator is significantly reduced. At this time, the MSIVs are also assumed to close. This is conservative because the a MSIV closure is expected to occur at approximately 15 seconds based on a low steam line pressure signal.

6. AFW flow begins at the time of the break at the runout flow of 1210 GPM.

7. A sink volume is maintained at a constant pressure of 14.7 psia.

8. Smooth commercial steel pipe, with a relative roughness of 0.00015ft is assumed for all piping.

9. A throat with an area of 1.38 ft^2 is assume to limit the mass and energy release from the SG side.

The feedwater line break analysis was performed using the RELAP5/MOD1 (Ref. 6.2.1.2-6) computer program. Assumptions used in this analysis include:

REFERENCES

Section 6.2:

6.2.1.1-1 Bechtel Power Corporation Computer Code: COPATTA User's Guide, Volume I, "Practical User's Guide", Volume II, "Theoretical User's Guide" 1974.

6.2.1.1-2 Bechtel Power Corporation, Topical Report No. BN-TOP-3, Rev. 4, "Performance and sizing of Dry Pressure Containments", March 1983.

6.2.1.1-3 Slughterbeck, D. C., Review of Heat Transfer Coefficients for Condensing Steam in a Containment Building Following a Loss of Coolant Accident, IN-1388, September 1970.

6.2.1.1-4 Uchida, H., A. Ogama, and Y. Togo, "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors", Proceedings of the Third International Conference on the Peaceful Uses of Atomic Energy, Volume 13, Session 3.9, United Nations, Geneva (1964).

6.2.1.1-5 Tagami, Takashi, "Interim Report on Safety Assessments and Facilities, Establishment Project in Japan for Period Ending June 1965 (No. 1)".

6.2.1.2-1 Bechtel Power Corporation, "COPDA Compartment Pressure Design Analysis", (Bechtel Computer Code), 1973.

6.2.1.2-2 Bechtel Power Corporation, "Subcompartment Pressure and Temperature Transient Analysis", Topical Report No. BN-TOP-4, (Rev. 1), October 1977.

6.2.1.2-3 Crane Co., "Flow of Fluids", Technical Paper No. 410, 1969.

6.2.1.2-4 Idel' Chik, I.E., "Handbook of Hydraulic Resistance Coeffcients of Local Resistance and of Friction", AEC-TR-6630, 1966.

6.2.1.2-5 Shepard, R.M., H.W. Massie, R.H. Mark and P.J. Doherty, "Westinghouse Mass and Energy Release Data for Containment Design", WCAP-8264-P-A, Proprietary (June 1975) and WCAP-8312-A Revision 1, Nonproprietary (June 1975).

6.2.1.2-6 RELAP 5/MOD1 Code manual Volume 1: System Models and Numerical Methods, NUREG/CR-1826, EGG-2070, 1980.

6.2.1.2-7 American Nuclear Standard, "Design Basis for Protection of Light Water Nuclear Power Plant Against Effects of Postulated Pipe Rupture", ANSI/ANS-58.2-1980.

Insert 6 →

INSERT 6

6.2.1.2-8 Peterson, C.E. et al, "RETRAN-03 MOD01 HLP-001, " developed by Computer Simulation and Analysis, Inc., for the Electric Power Research Institute, July, 1991.

TABLE 6.2.1.1-8

THERMOPHYSICAL PROPERTIES
OF STRUCTURAL HEAT SINKS
FOR LOCA AND MSLB ANALYSIS

Material	Thermal Conductivity (Btu/hr-ft-°F)	Volumetric Heat Capacity (Btu/ft^3-°F)
Amercote 90 (organic)	.0375 .375	49.9
Dimetcote 6 (Inorganic)	0.633	21.67
Nutech Paint	0.1258	28.29
Air	0.0174	0.0103
Carbon Steel	25.0	54.0
Concrete	0.8	30.0
Stainless Steel	9.4	54.0

TABLE 6.2.1.2-1 (Continued)

SHORT-TERM MASS AND ENERGY RELEASE RATES
FOR SUBCOMPARTMENT ANALYSES

J. MAIN STEAM LINE DOUBLE-ENDED GUILLOTINE BREAK
USED FOR MAIN STEAM LINE SUBCOMPARTMENT ANALYSIS
~~(Blowdown includes no arbitrary margin)~~

Time(s)	Mass Flow (lbm/sec)	Energy Flow (Btu/sec)	Avg Enthalpy (Btu/lbm)
0.0	0.0	0.0	0.0
0.0066	1.7400E+04	2.0880E+07	1200.0
0.01	1.2181E+04	1.4617E+07	1200.0
0.013	1.1527E+04	1.3832E+07	1200.0
0.025	1.1063E+04	1.3276E+07	1200.0
0.05	1.0454E+04	1.2545E+07	1200.0
0.1	9.7810E+03	1.17370E+07	1200.0
0.125	9.9450E+03	1.1934E+07	1200.0
0.15	1.0381E+04	1.2457E+07	1200.0
0.20	1.0072E+04	1.2086E+07	1200.0
0.25	9.7999E+03	1.1760E+07	1200.0
0.30	9.6181E+03	1.1542E+07	1200.0
0.35	9.5272E+03	1.1433E+07	1200.0
0.40	9.4363E+03	1.1324E+07	1200.0
0.45	9.4363E+03	1.1324E+07	1200.0
0.5	9.3454E+03	1.1214E+07	1200.0

THIS TABLE DELETED

* An enthalpy of 1200 Btu/lbm was assumed conservatively throughout the transient.

TABLE 6.2.1.2-13

MAIN STEAM LINE AND FEEDWATER LINE
SUBCOMPARTMENTS ANALYSIS

Node	Net Volume (ft^3)	Peak Differential Pressure (psid)	Time to Peak Differential Pressure (sec)	Design Pressure (psid)	Design Margin (%)
1	6,030.42	~~12.97~~ 13.03	0.019	30.15	~~132.4~~ 131.2
2	16,982.55	~~1.31~~ 1.12	0.055	15.00	**
3	5,332.80	~~13.65~~ 13.84	~~0.017~~ 0.018	30.15	~~120.9~~ 117.8
4	29,518.67	~~1.49~~ 1.43	~~0.053~~ 0.054	15.00	**
5	6,766.11	~~6.45~~ 5.06	~~0.029~~ 0.031	14.25	~~120.9~~ 181.6
6	15,748.10	~~1.63~~ 1.53	~~0.044~~ 0.045	15.00	**
7	5,958.95	~~6.44~~ 4.79	~~0.026~~ 0.030	14.25	~~121.3~~ 197.5
8	19,100.75	~~1.89~~ 1.76	~~0.044~~ 0.045	15.00	**
9	35,385.71	~~0.90~~ 0.80	~~0.050~~ 0.037	25.50	**
10	32,975.70	~~0.98~~ 0.90	~~0.050~~ 0.054	25.50	**
11	3.18 x 10^6	-	-	-	-
12	50,653.39	~~0.03~~ 0.06	~~0.08~~	negligible	N.A.
13	48,905.96	~~0.05~~ 0.08	~~0.09~~	negligible	N.A.
14	58,770.56	0.04 0.40	~~0.09~~	negligible	N.A.

** Large margin exists.

Initial conditions for all nodes are identical. Temp. - 120°F, Press. - 14.7 psia, and relative humidity - 50%

Replace with attached table

TABLE 6.2.1.2-14

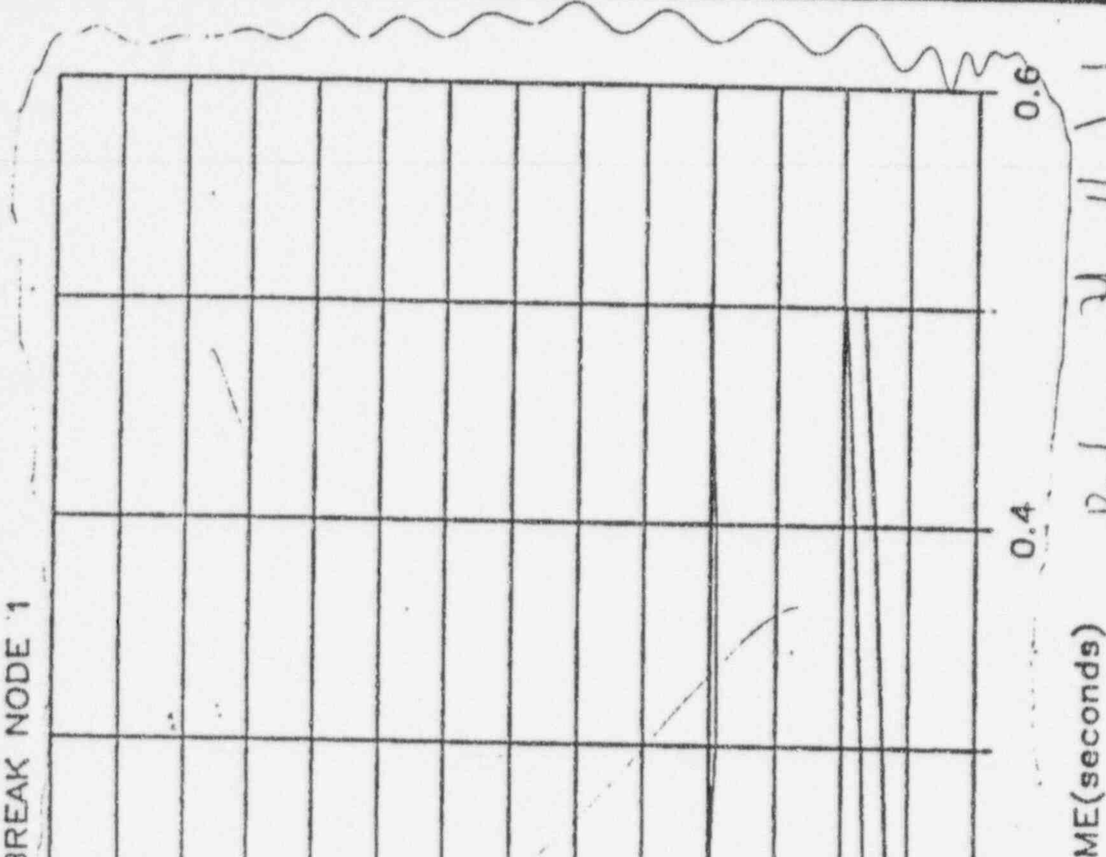
MAIN STEAM LINE AND FEEDWATER SUBCOMPARTMENT ANALYSIS
JUNCTION DESCRIPTION
~~STEAM LINE BREAK ANALYSIS~~

Nodes From	Nodes To	Vent Area (ft^2)	Length/ Area (ft^{-1})	Head Loss Coefficients $K_{Contraction}$	$K_{Expansion}$	$K_{Grating}$	K_{Total}	Flow Coefficient
1	2	154.92	0.23	0.1962	1.0	-	1.1962	0.91
1	3	68.85	0.23	0.365	1.0	-	1.365	0.85
1	5	204.43	0.025	0.3163	1.0	0.3	1.6163	0.78
2	6	555.83	0.012	0.2787	1.0	0.3	1.5787	0.79
2	11	539.79	0.0064	0.2787	1.0	0.3	1.5787	0.79
2	12	208.80	0.61	0.165	1.0	-	1.165	0.92
3	4	189.34	0.16	0.138	1.0	-	1.138	0.93
3	7	190.72	0.028	0.306	1.0	0.3	1.606	0.78
4	8	676.39	0.01	0.2791	1.0	0.3	1.5791	0.79
4	[illegible]	468.04	0.0052	0.2791	1.0	0.3	1.5791	0.79
4	12	36.00	0.76	0.453	1.0	-	1.453	0.83
5	6	179.22	0.22	0.196	1.0	-	1.196	0.91
5	7	71.65	0.19	0.365	1.0	-	1.365	0.85
5	9	204.43	0.17	0.316	1.0	0.3	1.616	0.78
6	9	555.83	0.0099	0.2787	1.0	0.3	1.5787	0.79
6	13	192.49	0.604	0.165	1.0	-	1.165	0.92
7	8	219.03	0.15	0.1378	1.0	-	1.1378	0.93
7	10	190.72	0.019	0.306	1.0	0.3	1.606	0.78
8	10	673.08	0.009	0.279	1.0	0.3	1.579	0.79
8	13	39.82	0.81	0.443	1.0	-	1.443	0.83
9	10	361.35	0.26	0.111	1.0	-	1.111	0.95
9	11	752.93	0.004	0.335	1.0	0.3	1.635	0.78
9	14	225.13	0.50	0.1837	1.0	0.3	1.1837	0.91
10	11	645.72	0.0043	0.348	1.0	0.3	1.648	0.77
10	14	65.70	0.665	0.425	1.0	0.3	1.425	0.83
12	11	796.85	0.002	0.34	1.0	0.3	1.64	0.78
12	13	1667.57	0.004	0.2862	1.0	0.3	1.586	0.79
13	14	403.35	0.004	0.448	1.0	0.3	1.748	0.75
14	11	755.98	0.0024	0.403	1.0	0.3	1.703	0.76

TABLE 6.2.1.2-14

MAIN STEAM LINE AND FEEDWATER SUBCOMPARTMENT ANALYSIS
JUNCTION DESCRIPTION
(STEAM LINE BREAK ANALYSIS)

Junction	Nodes From	To	Vent Area (ft^2)	Inertia Length (ft)	Loss Coeff.	Hydraulic Diameter (ft)	Friction Length
1	1	2	154.92	35.630	1.1962	4.4662	2'
2	1	3	68.85	15.840	1.3650	3.1864	8"
3	1	5	204.43	5.111	1.6163	0.1296	2"
4	2	6	555.83	6.670	1.5787	0.1296	2"
5	2	11	539.79	3.455	1.5787	0.1296	2"
6	2	12	77.16	47.070	1.3763	3.2540	2'
7	3	4	189.34	30.290	1.1380	4.4847	2'
8	3	7	190.72	5.340	1.6060	0.1296	2"
9	4	8	676.39	6.764	1.5791	0.1296	2"
10	4	11	468.04	2.434	1.5791	0.1296	2"
11	4	12	36.00	27.360	1.4530	1.0588	8"
12	5	6	136.80	30.096	1.2686	2.7959	2'
13	5	7	58.60	11.130	1.4007	2.4298	2'
14	5	9	204.43	3.475	1.6160	0.1296	2"
15	6	9	555.83	5.503	1.5787	0.1296	2"
16	6	13	192.49	116.300	1.1650	3.9342	8"
17	7	8	219.03	32.850	1.1378	5.7274	2'
18	7	10	190.72	3.624	1.6060	0.1296	2"
19	8	10	673.08	6.058	1.5790	0.1296	2"
20	8	13	39.82	32.250	1.4430	1.8776	8"
21	9	10	361.35	93.950	1.1110	3.5896	2'
22	9	11	752.93	3.012	1.6350	0.1296	2"
23	9	14	108.71	54.355	1.3473	5.39[illegible]7	2'
24	10	11	645.72	2.777	1.6480	0.1296	2"
25	10	14	65.70	43.690	1.4250	2.1366	8"
26	12	11	796.85	1.594	1.6400	0.1296	2"
27	12	13	1667.57	6.670	1.5860	0.1296	2"
28	13	14	403.35	1.613	1.7480	0.1296	2"
29	14	11	755.98	1.814	1.7030	0.1296	2"



Replace with attached Figure A

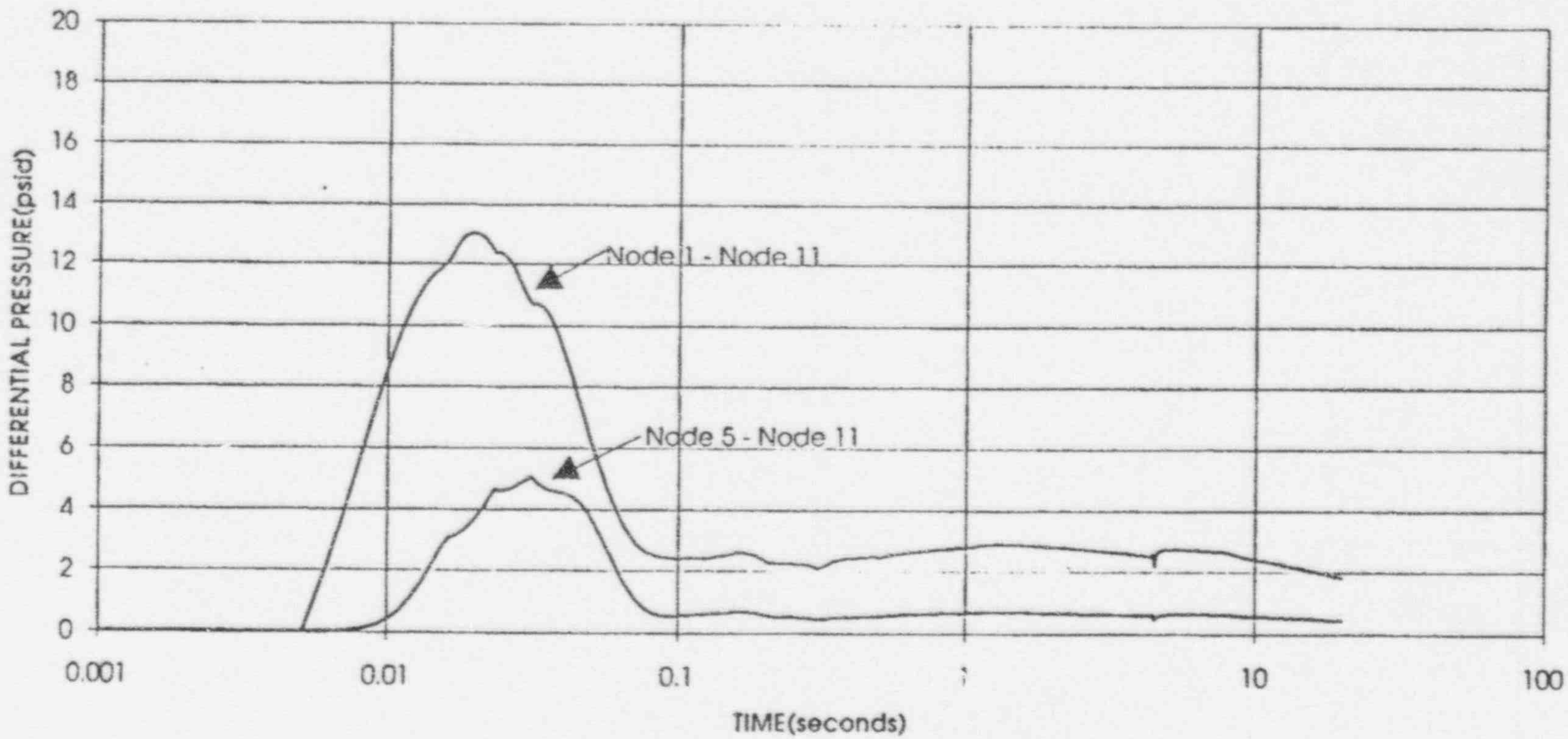
SOUTH TEXAS PROJECT
UNITS 1 & 2

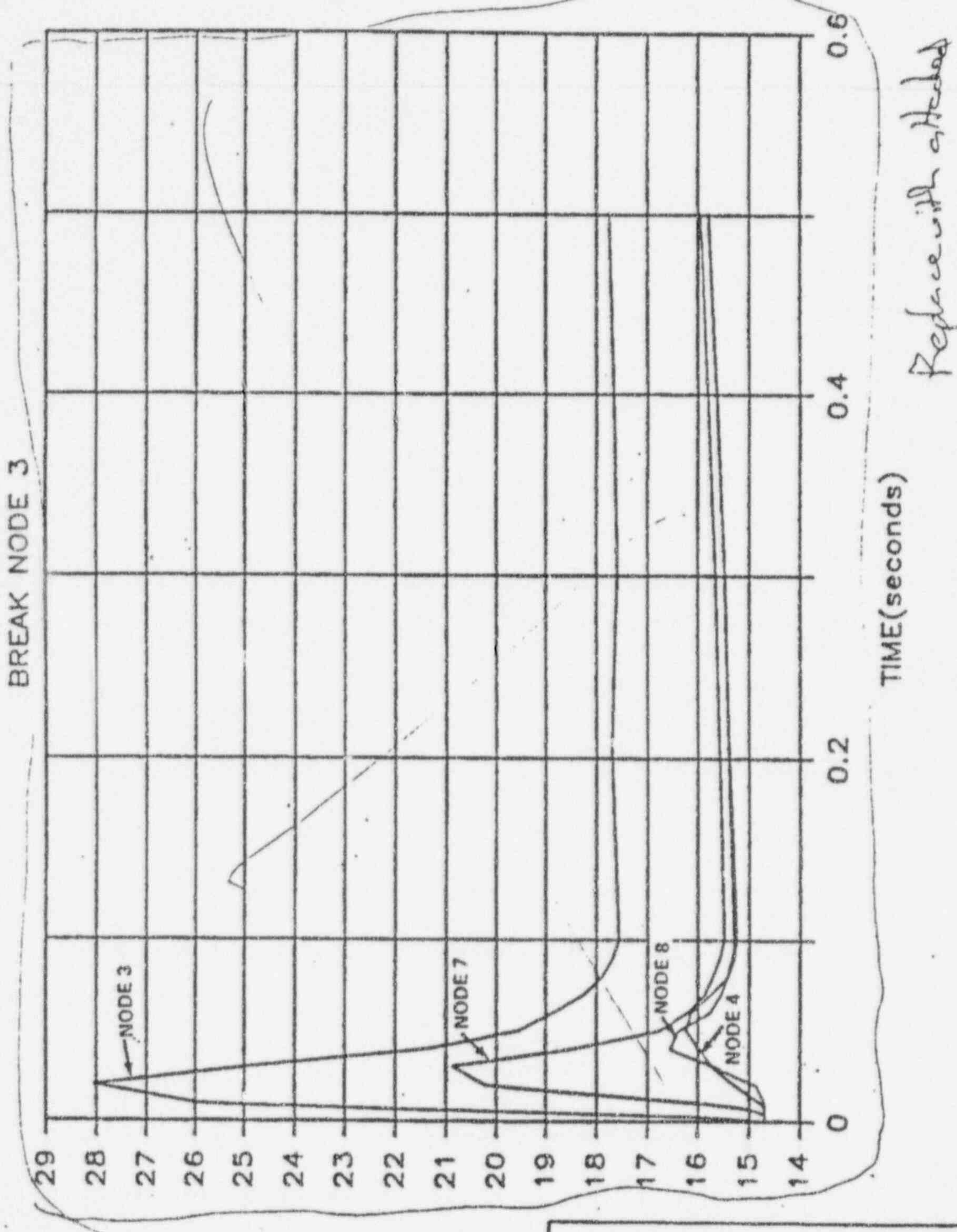
MSLB SUBCOMPARTMENT
P-T ANALYSIS
(Sheet 1 of 2)

Figure 6.2.1.2-25 Revision 0

Figure A

BREAK NODE 1





Replace with attached Figure B

SOUTH TEXAS PROJECT UNITS 1 & 2

MSLB SUBCOMPARTMENT
P-T ANALYSIS
(Sheet 2 of 2)

Figure 6.2.1.2-25

Revision 0

FIGURE B

BREAK NODE 3