Florida Power & Light, 9760 S.W. 344 St. Homestead, FL 33035

FPL.

AUG 0 5 2011

L-2011-233
10 CFR 50.90
10 CFR 2.390

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to NRC Request for Additional Information Regarding
Extended Power Uprate License Amendment Request No. 205 and
Reactor Systems Issues

References:

(1) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request No. 205: Extended Power Uprate (EPU)," (TAC Nos. ME4907 and ME4908), Accession No. ML103560169, October 21, 2010.

(2) Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU – Reactor Systems (SRXB) Request for Additional Information – Round 1.3 (Part 3)", July 21, 2011.

By letter L-2010-113 dated October 21, 2010 [Reference 1], Florida Power and Light Company (FPL) requested to amend Renewed Facility Operating Licenses DPR-31 and DPR-41 and revise the Turkey Point Units 3 and 4 Technical Specifications (TS). The proposed amendment will increase each unit's licensed core power level from 2300 megawatts thermal (MWt) to 2644 MWt and revise the Renewed Facility Operating Licenses and TS to support operation at this increased core thermal power level. This represents an approximate increase of 15% and is therefore considered an extended power uprate (EPU).

On June 30, 2011, a public meeting was held with the U.S. Nuclear Regulatory Commission (NRC) Project Manager (PM), applicable NRC technical reviewers, and FPL representatives to discuss proposed NRC requests for information (RAI) related to the EPU License Amendment Request (LAR). During the meeting, forty proposed RAI questions from the NRC Reactor Systems Branch (SRXB) on loss-of-coolant (LOCA) and non-LOCA safety analyses were discussed. On July 21, 2011, FPL received an email from the NRC PM containing the final RAI [Reference 2]. The RAI consisted of thirty-nine (39) of the questions previously discussed in the public meeting regarding non-LOCA, large break LOCA, and small break LOCA safety analyses. These RAI questions and the FPL responses to RAI questions SRXB-1.3.1-1.3.6 and 1.3.16-1.3.39 are documented in Attachments 1 and 2 to this letter. As discussed previously, FPL's responses to SRXB-1.3.7-1.3.15 will be provided via separate correspondence.

Attachment 3 contains the application for withholding the proprietary information contained in Attachment 2 from public disclosure. As Attachment 2 contains information proprietary to Westinghouse Electric Company, LLC (Westinghouse), it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis for which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of §2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations.

AD84
NRR

Correspondence with respect to the copyright or proprietary aspects of items in the response to the RAI questions in Attachment 2 of this letter or the supporting Westinghouse affidavit should reference CAW-11-3224 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, PA 16066.

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the State Designee of Florida.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2010-113 [Reference 1].

This submittal contains no new commitments and no revisions to existing commitments.

Should you have any questions regarding this submittal, please contact Mr. Robert J. Tomonto, Licensing Manager, at (305) 246-7327.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 5, 2011.

Very truly yours,

Eric McCartney for M. Kiley

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Attachments (3)
CDs for PM Only (2)

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Resident Inspector, Turkey Point Nuclear Plant
Mr. W. A. Passetti, Florida Department of Health (wo Attachment 2)

Turkey Point Units 3 and 4

RESPONSE TO NRC RAI REGARDING EPU LAR NO. 205
AND SRXB REACTOR SYSTEMS ISSUES

ATTACHMENT 3

Westinghouse Affidavit for Attachment 2
August 4, 2011

This coversheet plus 8 pages

Westinghouse

Westinghouse Electric Company
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 720-0754
e-mail: greshaja@westinghouse.com
Proj letter: FPL-11-194

CAW-11-3224

August 4, 2011

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: FPL-11-194 P-Attachment, "Turkey Point Units 3 and 4 – Response to NRC Informal Request for Additional Information (RAI) from the Reactor Systems Branch (SRXB) Related to Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 (TAC Nos. ME 4907 and ME 4908)" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-11-3224 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Florida Power and Light.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-11-3224, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance

Enclosures

CAW-11-3224

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J A Gresham

J. A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 4th day of August 2011

Cynthia Olesky

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Cynthia Olesky, Notary Public
Manor Boro, Westmoreland County
My Commission Expires July 16, 2014
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390; it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in FPL-11-194 P-Attachment, "Turkey Point Units 3 and 4 – Response to NRC Informal Request for Additional Information (RAI) from the Reactor Systems Branch (SRXB) Related to Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 (TAC Nos. ME 4907 and ME 4908)" (Proprietary) for submittal to the Commission, being transmitted by Florida Power and Light letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for use by Turkey Point Units 3 and 4 is expected to be applicable for other licensee submittals in response to certain NRC requirements for Extended Power Uprate submittals and may be used only for that purpose.

This information is part of that which will enable Westinghouse to:

(a) Provide input to the U.S. Nuclear Regulatory Commission for review of the Turkey Point EPU submittals.

(b) Provide results of customer specific calculations.

(c) Provide licensing support for customer submittals.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of the information to its customers for the purpose of meeting NRC requirements for licensing documentation associated with EPU submittals.

(b) Westinghouse can sell support and defense of the technology to its customer in licensing process.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar information and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

Turkey Point Units 3 and 4

RESPONSE TO NRC RAI REGARDING EPU LAR NO. 205
AND SRXB REACTOR SYSTEMS ISSUES
(NON-PROPRIETARY)

ATTACHMENT 1

TABLE OF CONTENTS

Response to Request for Additional Information

The following information is provided by Florida Power and Light Company (FPL) in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support License Amendment Request (LAR) 205, Extended Power Uprate (EPU), for Turkey Point Nuclear Plant (PTN) Units 3 and 4 that was submitted to the NRC by FPL via letter (L-2010-113) dated October 21, 2010 [Reference 1].

By email from the NRC Project Manager (PM) dated July 21, 2011 [Reference 2], additional information regarding reactor safety analysis issues was requested by the NRC staff in the Reactor Systems Branch (SRXB) to support the review of the EPU LAR [Reference 1]. The RAI consisted of thirty-nine (39) questions regarding loss-of-coolant accident (LOCA) and non-LOCA analyses. The RAI questions and the applicable FPL response are documented below.

Note that this attachment (Attachment 1) presents the non-proprietary version of the RAI response. Attachment 2 presents the proprietary version of the RAI response. Attachment 3 contains the application for withholding the proprietary information contained in this attachment from public disclosure. As Attachment 2 contains information proprietary to Westinghouse Electric Company, LLC (Westinghouse), it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis for which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of §2.390 of the Commission's regulations. The proprietary information contained in the responses to RAI questions SRXB-1.3.24, 1.3.28, 1.3.30, 1.3.34, 1.3.36, and 1.3.39 is enclosed in brackets and the justification is annotated by means of lower case letters (a) and (c) located as a superscript immediately following the brackets, i.e., [ ]^{a,c}.

General

SRXB-1.3.1 Provide a copy of the PTN General Design Criteria.

As noted in PTN Updated Final Safety Analysis Report (UFSAR), Section 1.3, the General Design Criteria (GDC) used during licensing of the Turkey Point Nuclear Plant predate those provided today in 10 CFR 50, Appendix A. The PTN GDCs were developed based on the 1967 Atomic Energy Commission (AEC) Proposed General Design Criteria [as amended by the Atomic Industrial Forum (AIF)] and are addressed throughout the UFSAR. The original GDCs proposed by the AEC were published for public comment on July 10, 1967 and the AIF amended version of those GDCs was subsequently published on October 2, 1967. Note that several of the PTN GDC commitments have since been changed by newer commitments including 10CFR50 Appendix R commitments in UFSAR Appendix 9.6A in lieu of PTN GDC-3 for Fire Protection, 10CFR50 Appendix A GDC-4 for Leak-Before-Break (LBB) considerations, 10CFR50 Appendix A GDC-19 in lieu of PTN GDC-11 for control room, and 10CFR50 Appendix A GDCs 64, 63, and 60 in lieu of PTN GDCs 17, 18, and 70 for monitoring and control of radioactive effluents.

PTN GDC-1, Quality Standards: "Those systems and components of reactor facilities which are essential to the prevention or the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required."

PTN GDC-2, Performance Standards: "Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design."

PTN GDC-3, Fire Protection: "A reactor facility shall be designed to ensure that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features."

NOTE: PTN GDC-3 was replaced by PTN commitment to 10 CFR50 Appendix R.

PTN GDC-4, Sharing of Systems: "Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public."

PTN GDC-5, Records Requirements: "The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public."

PTN GDC-6, Reactor Core Design: "The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated."

PTN GDC-7, Suppression of Power Oscillations: "The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed."

NOTE: No PTN GDC-8

PTN GDC-9, Reactor Coolant Pressure Boundary: "The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime."

PTN GDC-10, Reactor Containment: "Reactor containment shall be provided. The containment structure shall be designed (a) to sustain, without undue risk to the health and safety of the public, the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity, and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires, the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public."

PTN GDC-11, Control Room: "The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without excessive radiation exposures of personnel."

NOTE: PTN GDC-11 replaced by PTN commitment to 10 CFR 50 GDC-19.

PTN GDC-12, Instrumentation and Control Systems: "Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables."

PTN GDC-13, Fission Process Monitors and Controls: "Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core."

PTN GDC-14, Core Protection System: "Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits."

PTN GDC-15, Engineered Safety Features Protection Systems: "Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features."

PTN GDC-16, Monitoring Reactor Coolant Leakage: "Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary."

PTN GDC-17, Monitoring Radioactivity Releases: "Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive."

NOTE: PTN GDC-17 superseded by PTN commitment to 10 CFR 50, Appendix A GDC-64, Monitoring Radioactive Releases;

PTN GDC-18, Monitoring Fuel and Waste Storage: "Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels".

NOTE: PTN GDC-18 superseded by PTN commitment to 10 CFR 50, Appendix A GDC-63, Monitoring Fuel and Waste Storage.

PTN GDC-19, Protection System Reliability: "Protection systems shall be designed for high functional reliability and inservice testability necessary to avoid undue risk to the health and safety of the public."

PTN GDC-20, Protection System Redundancy and Independence: "Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served."

NOTE: No PTN GDC-21

NOTE: No PTN-GDC-22

PTN GDC-23, Protection Against Multiple Disability for Protection Systems: "The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function or shall be tolerable on some other basis."

NOTE: No PTN GDC-24

PTN GDC-25, Demonstration of Functional Operability of Protection Systems: "Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred."

PTN GDC-26, Protection Systems Failure Analysis Design: "The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced."

PTN GDC-27, Redundancy of Reactivity Control: "Two independent reactivity control systems, preferably of different principles, shall be provided."

PTN GDC-28, Reactivity Hot Shutdown Capability: "The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition."

PTN GDC-29, Reactivity Shutdown Capability: "One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn."

PTN GDC-30, Reactivity Holddown Capability: "The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public."

PTN GDC-31, Reactivity Control System Malfunction: "The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits."

PTN GDC-32, Maximum Reactivity Worth of Control Rods: "Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core."

PTN GDC-33, Reactor Coolant Pressure Boundary Capability: "The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic load imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition."

PTN GDC-34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention: "The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes."

NOTE: No PTN GDC-35

PTN GDC-36, Reactor Coolant Pressure Boundary Surveillance: "Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided."

PTN GDC-37, Engineered Safety Features Basis for Design: "Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary, assuming unobstructed discharge from both ends."

PTN GDC-38, Reliability and Testability of Engineered Safety Features: "All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public."

PTN GDC-39, Emergency Power for Engineered Safety Features: "Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each system."

PTN GDC-40, Missile Protection: "Adequate protection for those engineered safety features, the failure of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failure."

NOTE: For primary loop piping PTN GDC-40 superseded by PTN commitment to 10 CFR 50, Appendix A GDC-4, Environmental and Dynamic Effects Design Bases.

PTN GDC-41, Engineered Safety Features Performance Capability: "Engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public."

PTN GDC-42, Engineered Safety Features Components Capability: "Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a loss-of-coolant accident to the extent of causing undue risk to the health and safety of the public."

PTN GDC-43, Accident Aggravation Prevention: "Protection against any action of the engineered safety features which would accentuate significantly the adverse after-effects of a loss of normal cooling shall be provided."

PTN GDC-44, Emergency Core Cooling System Capability: "An emergency core cooling system with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty."

PTN GDC-45, Inspection of Emergency Core Cooling System: "Design provisions shall, where practical, be made to facilitate physical inspection of all parts of the Emergency Core Cooling System, including reactor vessel internals and water injection nozzles."

PTN GDC-46, Testing of Emergency Core Cooling System Components: "Design provisions shall be made so that components of the Emergency Core Cooling System can be tested periodically for operability and functional performance."

PTN GDC-47, Testing of Emergency Core Cooling System: "Capability shall be provided to test periodically the operability of the Emergency Core Cooling System up to a location as close to the core as is practical."

PTN GDC-48, Testing of Operational Sequence of Emergency Core Cooling System: "Capability shall be provided to test initially, under conditions as close as practical to design, the full operational sequence that would bring the Emergency Core Cooling System into action, including the transfer to alternate power sources."

PTN GDC-49, Reactor Containment Design Basis: "The reactor containment structure, including access openings and penetrations, and any necessary containment heat removal systems, shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss of coolant accident, including the calculated energy from metal water or other chemical reactions that could occur as a consequence of failure of any single active component in the emergency core cooling system, will not result in undue risk to the health and safety of the public."

PTN GDC-50, NDT Requirement for Containment Material (Category A): "The selection and use of containment materials shall be in accordance with applicable engineering codes."

NOTE: No PTN GDC-51

PTN GDC-52, Containment Heat Removal Systems: "Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component."

PTN GDC-53, Containment Isolation Valves: "Penetrations that require closure for the containment functions shall be protected by redundant valving and associated apparatus."

NOTE: No PTN GDC-54

NOTE: No PTN GDC-55

NOTE: No PTN GDC-56

NOTE: No PTN GDC-57

PTN GDC-58, Inspection of Containment Pressure-Reducing Systems: "Design provisions shall be made to extent practical to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as pumps, valves, spray nozzles, tanks and sumps."

PTN GDC-59, Testing of Containment Pressure-Reducing Systems Components: "The containment pressure-reducing systems shall be designed, to the extent practical so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance."

PTN GDC-60, Testing of Containment Spray Systems: "A capability shall be provided to the extent practical to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical."

PTN GDC-61, Testing of Operational Sequence of Containment Pressure-Reducing Systems: "A capability shall be provided to test initially under conditions as close as practical to the design and the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources."

PTN GDC-62, Inspection of Air Cleanup Systems: "Design provisions shall be made to the extent practical to facilitate physical inspection of all critical parts of containment air cleanup systems, such as ducts, filters, fans, and dampers."

PTN GDC-63, Testing of Air Cleanup Systems Components: "Design provisions shall be made to the extent practical so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance."

PTN GDC-64, Testing Air Cleanup Systems: "A capability shall be provided to the extent practical for onsite periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits."

PTN GDC-65, Testing of Operational Sequence of Air Cleanup Systems: "A capability shall be provided to test initially under conditions as close to design as practical, the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability."

PTN GDC-66, Prevention of Fuel Storage Criticality: "Criticality in the new and spent fuel storage pits shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls."

PTN GDC-67, Fuel and Waste Storage Decay Heat: "Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities and to waste storage tanks that could result in radioactivity release which could result in undue risk to the health and safety of the public."

PTN GDC-68, Fuel and Waste Storage Radiation Shielding: "Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities."

PTN GDC-69, Protection Against Radioactivity Release From Spent Fuel and Waste Storage: "Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity."

PTN GDC-70, Control of Releases of Radioactivity to the Environment: "The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control must be justified (a) on the basis of 10 CFR 20 requirements, for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence."

NOTE: PTN GDC-70 superseded by PTN commitment to 10 CFR 50, Appendix A GDC-60, Control of Releases of Radioactivity to the Environment.

2.8.4.2 Overpressure Protection during Power Operation

SRXB-1.3.2 NUREG-0800 §5.2.2 specifies the acceptance criteria to be applied in reviews of analyses of overpressure protection during power operation for PWRs. Acceptance criterion 3.B.iii requires that the second safety-grade signal from the reactor protection system initiate the reactor scram. The licensee refers to the UFSAR Chapter 14 loss of load analysis to demonstrate that adequate overpressure protection exists in the Turkey Point units. This is not acceptable since the UFSAR Chapter 14 loss of load analysis is based upon the reactor tripping upon receipt of the first safety-grade signal from the reactor protection system.

a. Provide an overpressure protection analysis that meets the acceptance criteria for PWRs; specified in NUREG-0800 §5.2.2.

The Turkey Point units were licensed before the Standard Review Plan (SRP) [Reference 3] was issued. In the absence of SRP 5.2.2 or equivalent, adequate overpressure protection for the Turkey Point units is demonstrated by the UFSAR safety analyses (i.e., the Loss of Load and Turbine Trip analyses), which are based upon reactor scrams that are demanded by the first safety-grade signals received from the reactor protection system. Analyses of the Loss of Load and Turbine Trip event are therefore presented in LR Section 2.8.5.2.1 to show that the plant has adequate overpressure protection when operating at the proposed EPU power level. The results indicate that, after some necessary setpoint adjustments, adequate overpressure margin is maintained at the proposed EPU power level.

	EPU Analysis	Previous Analysis	Limit
Peak RCS Pressure (psia)	2746.6	2748.4	2748.5
Peak MSS Pressure (psia)	1197.07	1208.0	1208.5

This approach is consistent with the guidance of RS-001, which states, "The staff does not intend to impose the criteria and/or guidance in this review standard on plants whose design bases do not include these criteria and/or guidance. No backfitting is intended or approved in connection with the issuance of this review standard." This approach was accepted previously for the Point Beach Extended Power Uprate [Reference 4]. The Point Beach and Turkey Point units were both licensed before the SRP was issued.

b. Review the results of the new overpressure protection analysis, and revise, if necessary, the safety and relief valve setpoints that are required in order to provide adequate overpressure protection for operation under EPU conditions.

See Response a above. No additional changes are required to the safety and relief valve setpoints.

2.8.5.1.1 Reduction In Feedwater Enthalpy, Increase In Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve

SRXB-1.3.3 Table 2.8.5.1.1.2.2-1 indicates that the reactor is tripped from a turbine trip signal, at 42.9 seconds. Identify any and all other instances, among the accident analyses that have been submitted to support this EPU, of tripping the reactor from the turbine trip.

The Increase in Feedwater Flow event is the only accident analysis that models a reactor trip on turbine trip. If this trip is not modeled, then following turbine trip and feedwater isolation on high-high steam generator level, the transient will resemble a loss of normal feedwater (an RCS heatup event) with level dropping until a reactor trip occurs on a low-low steam generator level signal. As stated in LR Section 2.8.5.1.1.2.2.5, the resultant loss of normal feedwater event would be less limiting than the loss of feedwater event documented in LR Section 2.8.5.2.3.

SRXB-1.3.4 Correct the statement, on page 2.8.5.1.1-6, pertaining to the initial water level in "all four steam generators". The Turkey Point units are three-loop plants.

The statement on page 2.8.5.1.1-6 is incorrect. It should read, "...in all three steam generators..."

SRXB-1.3.5 The excessive feedwater flow event is ended by automatic closure of all feedwater control and isolation valves, closure of all feedwater bypass valves, a trip of the feedwater pumps, and a turbine trip on high-high steam generator water level. Identify the single failure that is assumed, and its effect, if any.

The most limiting single failure is the failure of one train of the reactor protection system. The other train of the protection system, which remains functional, carries out the protection functions and therefore, there is no effect.

SRXB-1.3.6 Explain how the hot zero power case (HZP) would be less limiting than the HZP steam line break case. The cooldown rate would be slower; but the reactor protection system response would be different (e.g., no high steam flow rate signal).

For Turkey Point, the RCS cooldown and subsequent depressurization induced by the HZP steam line break are much more adverse than the HZP feedwater malfunction. During a steam line break, the increased steam flow from the steam generators causes a higher heat extraction rate from the RCS than is seen from a feedwater malfunction. Furthermore, the HZP feedwater malfunction has a shorter duration than the HZP steam line break. Whereas the feedwater malfunction is terminated by a feedwater isolation signal, the steam line break is not fully terminated until the faulted steam generator blows dry since the break is modeled in an unisolatable location. The more adverse cooldown and depressurization for the steam line break result in more positive reactivity being added to the core, which degrades the shutdown margin causing a higher return to power. A comparison of the transient statepoints from the two events analyzed for the Turkey Point EPU confirms that the steam line break produces a significantly higher power level at a lower RCS pressure. Thus, it was concluded that the HZP steam line break is more limiting than the HZP feedwater malfunction.

2.8.5.1.2 Steam System Piping Failures Inside and Outside Containment

SRXB-1.3.7 **UFSAR §1.3.7 states, "For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the Safety Injection System adds shutdown reactivity so that with a stuck rod, no off-site power and minimum engineered safety features, there is no consequential damage to the fuel or the primary system and the core remains in place and intact."**

a. **LR §2.8.5.1.2.2.1.2 states that, "Cases were analyzed both with offsite power available (full coolant flow is maintained) and with a coincident loss of offsite power (causing the reactor coolant pumps to coast down 3.0 seconds following the break)." Provide the results of the analysis of the major rupture of a steam pipe at HZP, without offsite power, to show that the above design criterion is satisfied. Include a sequence of events table indicating the times and values of peak heat flux and minimum DNBR, and transient plots that are comparable to the reported results of the equivalent steamline break case with offsite power. Include a transient plot of core flow.**

   Response will be provided later under separate submittal.

b. **Provide the results of an analysis or evaluation of the major rupture of a steam pipe, at HFP, without offsite power, to show that there is no consequential damage to the fuel or the primary system before the reactor is tripped. If applicable, include a sequence of events table indicating the times and values of peak heat flux and minimum DNBR, and transient plots that are comparable to the reported results of the equivalent steamline break case with offsite power.**

   Response will be provided later under separate submittal.

SRXB-1.3.8 **During a major steam line rupture, trip of the main steam isolation valves (MSIVs) and safety injection (SI) system actuation would occur when high steam flow is detected coincident with either low reactor coolant system (RCS) average temperature or low steam line pressure. Actuation of the SI system could also occur when pressurizer low-pressure, or high containment pressure, or high differential pressure between the steam line header and any steam line is detected.**

During a credible steam line break, the high steam flow condition would not be reached. Consequently, the MSIVs would not be tripped, and safety injection would not be actuated. Safety injection would be actuated much later, by low pressurizer pressure. In either situation, SI would not be delivered until after the RCS depressurizes to below the SI pump shut off head.

a. **The credible break, with no safety injection or steamline isolation will generate power to match the steam release through the break (i.e., the opening of the largest steam system valve, or about 10%). The peak post-trip power level, for the 1.4 ft2 break, is 13%. Therefore, the credible break is bounded, regardless of the effect of different protection system logic schemes and setpoints. Verify that there is no steam system valve that can relieve more than 13% of nominal steam flow. Explain how the major steam line rupture can be said to bound the credible break when the two events rely upon different protection system actuation logic schemes and response times.**

   Response will be provided later under separate submittal.

b. **Provide the analysis results of a credible steam line break, including a time sequence of events table listing values for peak heat flux and minimum DNBR (if applicable) as well as their times of occurrence.. (If question 1 (above) is answered, then this question is withdrawn.)**

   Response will be provided later under separate submittal.

SRXB-1.3.9 Figure 2.8.5.1.2.2.1-7 depicts the steam generator shell-side mass transient for the faulted and intact loops. Flow from the main feedwater system, which is assumed to be in operation when the plant is at hot zero power (HZP) conditions, and from the auxiliary feedwater system, do not allow the steam generator shell side inventory to drop below about 100,000 lbs. By ten minutes, the steam generator shell side inventory is increasing, due to continued addition of auxiliary feedwater. Describe the procedures and/or trips and/or alarms that would be used by the operator to end the auxiliary feedwater flow at ten minutes. Verify that this action will be accomplished in ten minutes.

Response will be provided later under separate submittal.

SRXB-1.3.10 Explain the saw tooth shape of the curves in Figures 2.8.5.1.2.2.1-1 and 2.8.5.1.2.2.1-2. If the saw tooth curve shape is due to the size of the time step, used in the analysis, show that reducing the time step does not materially change the results or conclusions of the analysis.

Response will be provided later under separate submittal.

SRXB-1.3.11 Table 2.8.5.1.2.2.1-1 indicates the core becomes subcritical at 186.25 seconds. Why is nuclear power still being generated, at a rate greater than 3%, more than six minutes after the core becomes subcritical?

Response will be provided later under separate submittal.

SRXB-1.3.12 In Figure 2.8.5.1.2.2.2-4, what is steam break flow, and how does it differ from the faulted loop SG outlet steam flow?

Response will be provided later under separate submittal.

SRXB-1.3.13 Describe, physically, the SG outlet steam flow (about 400 lbm/sec) after the reactor trip in Figure 2.8.5.1.2.2.2-4.

Response will be provided later under separate submittal.

SRXB-1.3.14 The Turkey Point units have four high head safety injection (HHSI) pumps shared between both units and all four receive the SI signal and begin delivering flow. PTN GDC-4, Sharing of Systems states: "Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public."

a. **Explain how the sharing of HHSI pumps between the two units meets the requirements of PTN GDC-4.**

   Response will be provided later under separate submittal.

b. **If an SI signal is generated in one unit, can it lead to the shutdown of both units?**

   Response will be provided later under separate submittal.

c. **If an SI signal is generated in one unit, what is the destination of the SI flow that is pumped in the other unit?**

   Response will be provided later under separate submittal.

d. **Why are the HHSI pumps assumed to be operating on degraded performance curves? Provide the degraded performance curves. Compare the degraded performance curves to the design performance curves.**

   Response will be provided later under separate submittal.

SRXB-1.3.15 If a steam line break were to occur at a location inside containment, at HFP conditions, how would the resulting adverse environment affect the generation of an overpower ΔT reactor trip signal?

Response will be provided later under separate submittal.

2.8.5.1.2(J) Minor steam line breaks (< 1.4 ft2) are said to be bounded by the major steam line break. Show that there is no minor break, larger than a credible break; but too small to cause steam line isolation, that is not bounded by the major steam line break.

Response will be provided later under separate submittal.

2.8.5.2.1 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam Pressure Regulator Failure

SRXB-1.3.16 One of the acceptance criteria for this event is to show that it will not generate a more serious plant condition without the occurrence of another, independent fault. Explain and document the rationale which concludes that this criterion is satisfied by verifying that no water relief through the PSVs occurs.

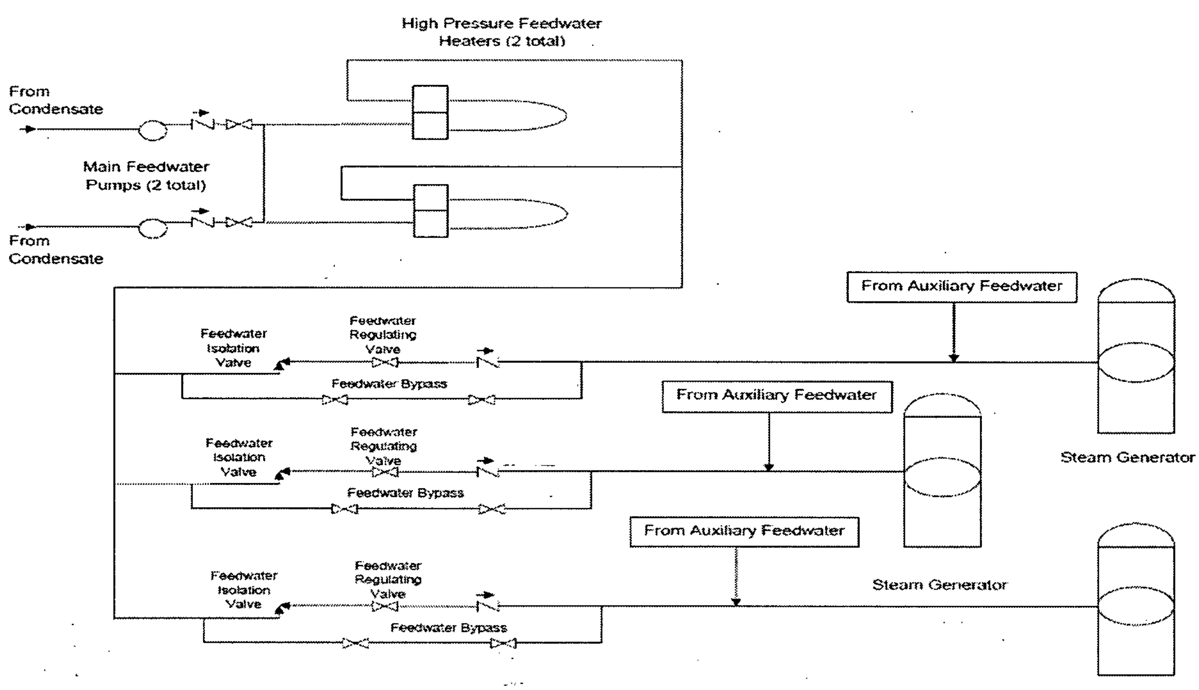
It can be postulated that a Loss of Load/Turbine Trip (LOL/TT) event could generate a more serious plant condition if fuel failure occurs, primary side overpressurization occurs, secondary side overpressurization occurs, or if the primary side pressure boundary is damaged resulting in an uncontrolled loss of inventory. Separate acceptance criteria are used to show that system over-pressurization and fuel damage will not occur. However, even without over-pressurization, in the worst case scenario the expansion of the reactor coolant could cause the pressurizer to become water solid thus requiring the PSVs to pass liquid. This could possibly result in damage to the RCS pressure boundary if the valve can no longer relieve pressure adequately, or lead to an uncontrolled loss of inventory if the valve is unable to close properly. This would be considered a small-break LOCA accident, which is a Condition III event. Therefore, by demonstrating that the pressurizer does not fill and that no water relief occurs out of the PSVs, it is confirmed that this event will not generate a more serious plant condition.

It is noted that the LOL/TT analysis assumptions are aimed at maximizing the RCS heatup and pressure, which occur quickly (usually within the first 15 seconds of the event). For this reason, the LOL/TT transient is terminated after 100 seconds. Therefore, for the period of time considered, pressurizer level increase occurs but not to the point where filling is a concern and the acceptance criterion, with respect to generating a more serious plant condition, is met. The long-term effects of a Loss of Load / Turbine Trip event, including pressurizer overfill, are bounded by the Loss of Normal Feedwater / Loss of AC Power event.

2.8.5.2.2 Loss of Non-Emergency AC Power to the Station Auxiliaries

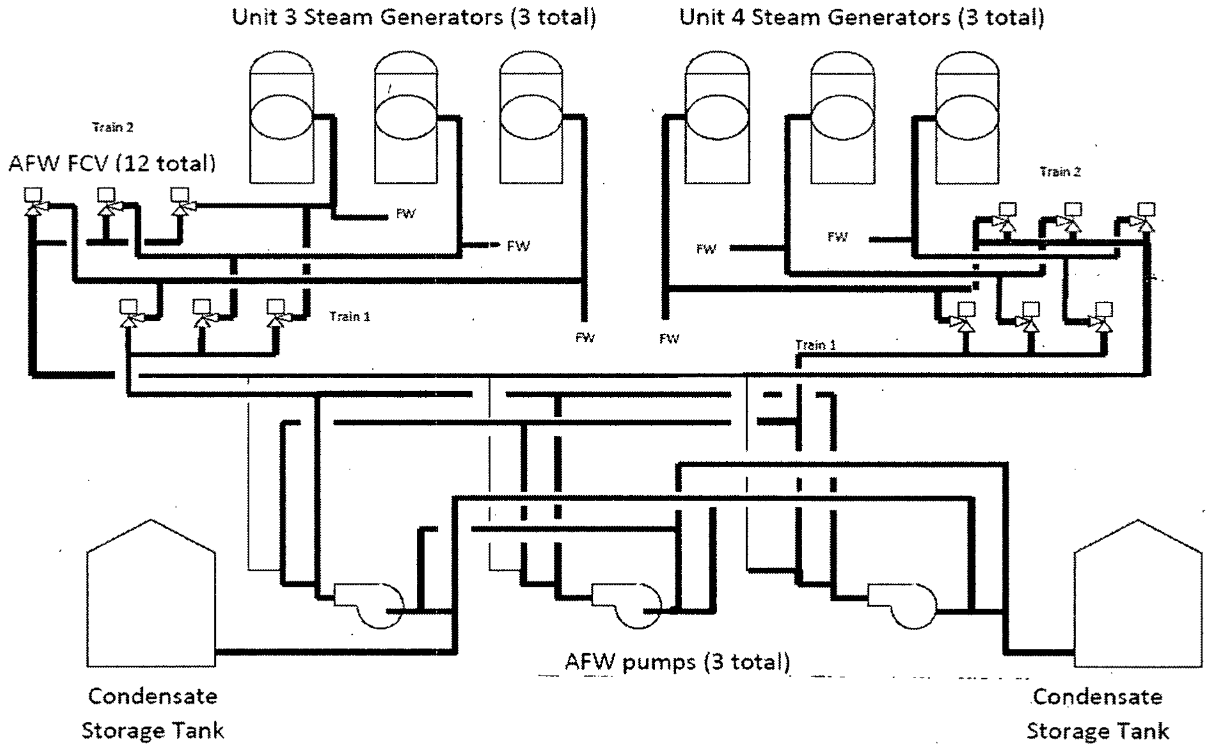
SRXB-1.3.17 Provide flow diagrams of the main and auxiliary feedwater systems.

Figure 1.3.17-1: Current Unit 3/4 Main Feedwater Flow Path Sketch



Note 1: Figure 1.3.17-1 shows major components used for Main Feedwater flow control and does not include items such as manual valves, specialty components, and instrumentation.
Note 2: For EPU, flow control modifications to the Main Feedwater lines, such as upgrading to fast acting Feedwater Isolation Valves and additional Feedwater Bypass Isolation Valves, will not affect the AFW flow path.

Figure 1.3.17-2: Current Auxiliary Feedwater Flow Path Sketch



Note 1: Figure 1.3.17-2 shows major components used for AFW flow control and does not include items such as manual valves, specialty components, and instrumentation.
Note 2: EPU does not impact the AFW flow path.

SRXB-1.3.18 If the three turbine-driven auxiliary feedwater pumps, shared by Units 3 and 4, were to be started:

a. Explain how the flow is limited to the affected unit's steam generators.

Auxiliary feedwater flow is limited to the steam generators using a system of air operated control valves (see Figure 1.3.18-2). The flow through the Flow Control Valves (FCVs) is controlled by the valve position. The valve positions are controlled automatically to maintain flow at the value pre-set by the operator at the hand indicating controller in the control room.

b. Describe the response and/or status of the unit in which the auxiliary feedwater start signal was not generated.

For the non-accident unit, no AFW signal is generated, therefore the associated FCVs remain closed while the affected unit's FCVs open. This ensures flow is provided to only the unit which generated the auxiliary feedwater start signal.

Following a transient in on the accident unit, the Reactor Protection System / Engineered Safety Feature Actuation System (RPS/ESFAS) automatically initiates the AFW System by opening the affected unit's steam inlet valves to all three turbines. Position switches on the steam inlet valves activate solenoid valves for that unit's FCVs which initiates AFW system flow by opening the affected unit's FCVs. The unaffected unit's normally closed FCVs remain closed since their steam supply valves did not open.

SRXB-1.3.19 Explain how the sharing of three turbine-driven auxiliary feedwater pumps between the two units meets the requirements of PTN GDC-4.

The AFW System is shared between Units 3 and 4 with three turbine driven AFW pumps that can draw water from either unit's CST and can feed the steam generators of either unit. PTN UFSAR Appendix A, Table A-1 presents a functional evaluation of the system components which are shared by the two units and concludes that the sharing of these system components do not result in undue risk to the health and safety of the public.

Currently, the minimum number of pumps required to meet the maximum demand is one AFW pump. Therefore, under emergency conditions, the AFW System can tolerate the failure of one of the operating AFW pumps while the third is out of service and still meet its design requirements.

After EPU, a single AFW pump will continue to meet the accident flow requirements of both units. As described in LR Section 2.5.4.5.2.5, this will be achieved by refurbishing AFW pumps B, C and the spare to meet performance requirements; as well as removing the travel stops from the AFW FCVs allowing them to fully open. Therefore, PTN GDC-4 will continue to be met after EPU.

2.8.5.2.3 Loss of Normal Feedwater Flow

SRXB-1.3.20 Verify that the decay heat level, predicted by the ANSI/ANS-5.1-1979 decay heat model, includes a 2σ adder to account for uncertainties.

As documented in Section 4.5 of WCAP-14882, the Westinghouse RETRAN model utilizes a built-in ANS-1979 decay heat model inclusive of a 2-sigma uncertainty. Since the Loss of Normal Feedwater Flow event was analyzed using the Westinghouse RETRAN computer code, the 2-sigma uncertainty was included in the decay heat model.

2.8.5.4.5 Chemical and Volume Control System Malfunction

SRXB-1.3.21 SRP Section 15.4.6 [1] lists the following acceptance criteria for B dilution event analyses:

> ***If operator action is required to terminate the transient, the following minimum time intervals must be available between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost:***
>
> ***A. During refueling: 30 minutes.***
>
> ***B. During startup, cold shutdown, hot shutdown, hot standby, and power operation: 15 minutes.***

The applicant's analysis of the Chemical and Volume Control System Malfunction addresses only Modes 1, 2 and 6.

a. **Provide analyses for this event in Modes 3, 4, and 5 (hot standby, hot shutdown, and cold shutdown, respectively). Initial conditions should consider the available shutdown margin, RCS pressure and charging flow, control rod positions and operability, available instrumentation and protective functions, and active RCS water volume (e.g., mid-loop operation) that are appropriate to each of these Modes.**

The Turkey Point units were licensed before the Standard Review Plan (SRP) was issued. Turkey Point Units 3 and 4 were licensed to the requirements in Regulatory Guide 1.70, Revisions 0 and 1 which require explicit Boron Dilution calculations in Modes 1, 2 and 6. Subsequent revisions to Regulatory Guide 1.70 and the Standard Review Plan have added requirements to consider boron dilutions in all six operating modes.

In January of 1985 in response to Generic Issue 22 which dealt with Boron Dilution in lower modes, the NRC issued Generic Letter (GL) 85-05 with the subject: "Inadvertent Boron Dilution Events". GL 85-05 states that "the consequences are not severe enough to jeopardize the health and safety of the public and do not warrant backfitting requirements for boron dilution events at operating reactors." The Turkey Point Units 3 and 4 UFSAR contains an explicit Boron Dilution calculation for Modes 1, 2 and 6 consistent with Regulatory Guide (RG) 1.70, Revisions 0 and 1. The NRC position is further supported by the letter from S. H. Hanauer (NRR) to R. J. Mattson (NRR) entitled: "inadvertent Boron Dilution", dated: March 10. 1982, where it is stated that the NRC concludes that there is inadequate justification to require all licensees to meet the SRP criteria.

b. **List the trips, alarms and other indications, which are required to be operable in each Mode, and which could alert the operator to an abnormal situation.**

Trips, alarms and other indications that could alert an operator that a dilution transient is occurring include those listed below. Each of these is required to be operable in the Modes indicated below by Technical Specifications (TS).

a. Indicated increase in Source Range Neutron Flux count rate (Modes 3, 4, 5, 6)
b. Source Range reactor trip (Modes 2, 3, 4, 5)
c. Intermediate Range reactor trip (Modes 1 and 2)
d. Axial-Flux-Difference Alarm (Modes 1 and 2)
e. Control rod insertion limit low and low-low alarms (Modes 1 & 2)
f. Overtemperature ΔT reactor trip (Modes 1 & 2)
g. Power Flux, high and low reactor trips (Modes 1 & 2)

The mitigation of an inadvertent boron dilution event due to CVCS malfunction may be manually or automatically initiated. In Mode 1, two Intermediate Range Neutron Flux Monitors and three Power Range Neutron Flux Monitors and Overtemperature ΔT are required (TS 3.3.1). In Mode 2, two Source Range Neutron Flux Monitors, two Intermediate Range Neutron Flux Monitors, three Power Range Neutron Flux Monitors and Overtemperature ΔT are required (TS 3.3.1). In Mode 6, one primary Source Range Neutron Flux Monitor with continuous visual indication in the control room and audible indication in the containment and control room is required. A backup monitor with continuous visual indication in the control room is also required (TS 3.9.2).

c. **Identify the trip, alarm or other indication that is assumed to alert the operator to the possibility that a B dilution event is occurring.**

In the analysis, no specific trip, alarm or indication is explicitly assumed to alert the operator consistent with the requirements of RG 1.70, Revisions 0 and 1. Instead it is recognized that one or more trips, alarms, or indications listed in the response to Item b above would alert the operator.

d. **Show that the operator, working according to the applicable procedures, will locate the B dilution source and flow path(s), and terminate the dilution flow within 15 minutes after receipt of the assumed trip, alarm, or other indication.**

The Turkey Point licensing basis requires that there be 15 minutes in Modes 1 and 2 and 30 minutes in Mode 6 from the start of the dilution to a loss of shutdown margin consistent with RG 1.70, Revisions 0 and 1.

As stated in LR Section 2.8.5.4.5.2.3, in Mode 1 the plant can be operated in either automatic or manual rod control. With the reactor in automatic rod control, the power and temperature increase from the boron dilution results in insertion of the control rods and a decrease in available shutdown margin. The rod insertion resulting from the dilution would reach the rod insertion limit alarms (low and low-low) which would serve to alert the operator to the dilution event. With the reactor in manual rod control, the power and temperature rise from the boron dilution will cause the reactor to reach the power range high neutron flux trip setpoint or the OTΔT trip setpoint, resulting in a reactor trip. The operator is alerted to the boron dilution by the reactor trip.

In Mode 2, the reactor is in manual rod control. The power and temperature rise from the boron dilution will cause the reactor to reach either the source range trip setpoint if below P-6 or the power range high neutron flux trip low setpoint, resulting in a reactor trip.

In Mode 6, the operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the containment building and the control room.

In addition to the trips, alarms and indications discussed in response to question b above, operators have audible indication of primary water flow in all modes of operation, audible indication of control rod motion, and audible indication of source range neutron count rates when in modes 3, 4, 5, 6 and Refueling. Any unexpected trip, alarm, or indication listed would prompt an operating crew to analyze the indication or alarm, respond in accordance with approved annunciator response procedures (in the event of an unexpected alarm), or in accordance with approved Off Normal or Emergency Operating Procedures designed to diagnose and mitigate primary water flow path malfunctions prior to losing shutdown margin.

Operators are trained in the accredited Initial Licensed Operator Training Program that includes classroom and simulator training in applicable procedures for

- malfunction of the boron concentration control system,
- loss of boration flowpaths,
- potential dilution of reactor coolant loops, and
- shutdown margin.

The accredited Licensed Operator Continuing Training Program includes refresher training on the same subjects and in the same settings on a periodic basis.

As noted in UFSAR Chapter 14.1.5, Chemical and Volume Control System Malfunction,

> "...if an unintentional dilution of boron in the reactor coolant system does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost."

Evaluations of operator performance in real time simulator scenarios and individual Job Performance Measures indicate that actions to mitigate inadvertent boron dilution events consistently occur within the 15 and 30 minutes required for these time critical actions.

2.8.5.3.2 Single Reactor Coolant Pump Locked Rotor

SRXB-1.3.22 Provide how the acceptance criteria, "maximum cladding temperature" at the core hot spot remains below 2700°F stated in section 2.8.5.3.2.2.2 of the extended power uprate (EPU) licensing report (LR) was derived.

Although the NRC Final Acceptance Criterion of 1974 limits maximum clad temperature to 2200°F for the loss of coolant accident, the limit for the locked rotor accident is 2700°F. NS-NRC-89-3466 [Reference 5] contains a detailed summary of the technical and licensing bases for the use of the 2700°F peak clad temperature limit as an acceptable criterion for coolability in non-LOCA events. Appendix E of the Westinghouse testimony in the matter of Acceptance Criteria of Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors (Docket No. RM50-1) [Reference 6], demonstrates that clad integrity is maintained at clad temperatures as high as 2700°F, even under conditions of severe thermal shock (quenching) for a local metal-water reaction of less than 16 percent. In the locked rotor accident, there is no thermal shock due to quench. Therefore, experimental results form a conservative basis for the selection of 2700°F as the maximum clad temperature limit for the locked rotor accident. This conservatively ensures that the core will remain in place and geometrically intact with no loss of core cooling capability. This criterion is more limiting than those stated previously and is adopted for convenience of interpreting the results of this study.

2.8.5.4.1 Uncontrolled Rod Cluster Control Assembly Withdrawal from a Subcritical or Low-Power Startup Condition

SRXB-1.3.23 The analysis assumes two of the three reactor coolant pumps (RCPs) to be in operation. This is conservative with respect to the departure from nucleate boiling (DNB) transient. Explain the basis for this assumption, specifically addressing the following items:

a. Why is two out of three a more limiting or appropriate RCP configuration than other conceivable configurations, such as under full reactor coolant flow or one out of three RCPs?

For the analysis of the Uncontrolled Rod Cluster Control Assembly Withdrawal from a Subcritical or Low-Power Startup Condition (RWFS) event, a minimum core flow is conservative with respect to the calculated results of minimum departure from nucleate boiling ratio (DNBR) and peak fuel centerline temperature. For a given plant, the RCP configuration applied in the RWFS analysis is typically based on the Technical Specification requirement for the number of loops in operation when the reactor trip breakers are closed in Mode 3. For Turkey Point, although Technical Specification 3.4.1.2 requires that all three of the reactor coolant loops shall be operable with all reactor coolant loops in operation when the reactor trip breakers are closed in Mode 3, only two RCPs, which conservatively minimizes the core flow, were credited in the RWFS analysis. Note that with two of three RCPs providing flow, reverse flow in the inactive loop results in a total vessel flow that is less than two-thirds of full flow.

b. What analytic treatment of the local hot bundle flow is provided to ensure that the local conditions appropriately capture the degraded flow conditions?

In addition to the conservative degraded inlet flow to the local hot bundle indicated in response to SRXB-1.3.23a, a flow reduction to the hot assembly is included in the DNB analysis as described in WCAP-14565 [Reference 7].

SRXB-1.3.24 Provide the location of minimum DNB ratio (DNBR).

Minimum DNBR Axial Locations are defined from the bottom of the bottom nozzle.

Fuel Type	Minimum DNBR Axial Location
Upgrade Fuel	[ ] a,c inches
DRFA Fuel	[ ] a,c inches

SRXB-1.3.25 Provide additional details concerning the VIPRE analysis:

a. **Explain whether the VIPRE analysis considers only the time of minimum DNBR or if additional DNBRs are determined.**

   VIPRE analysis only considers the time of minimum DNBR.

b. **Explain how the time of minimum DNBR is determined if not through explicit DNB calculations.**

   Due to the relatively rapid nature of the RWFS transient, the core coolant conditions of temperature and pressure do not change significantly during the RWFS transient, and the core coolant flow remains constant. Therefore, the time of minimum DNBR is defined by the time at which the peak core heat flux occurs.

2.8.5.4.2 Uncontrolled Rod Cluster Control Assembly Withdrawal at Power

SRXB-1.3.26 The revised thermal design procedure (RTDP) cases consider reactivity insertion rates as high as 80 percent millirho (pcm) per second, whereas the standard thermal design procedure (STDP) cases consider significantly lower reactivity insertion rates. Explain and justify the discrepancy.

Two different criterion are evaluated for the Uncontrolled Rod Cluster Control Assembly Withdrawal at Power. The first criterion evaluated is the DNBR performance to the event and utilized the RTDP methodology. The second criterion evaluated is RCS peak pressure during the event and used the STDP methodology. Acceptable results could not be obtained for the STDP cases (i.e., those that demonstrate that the RCS pressure criterion is met) using high reactivity insertion rates. Sensitivity analysis showed that acceptable results could be obtained using a maximum insertion rate of 29 pcm/sec. For completeness, all the reactivity insertion rates utilized for the DNBR event (RTDP) were also presented even though some of the insertion rates are above the EPU allowable limit.

The limiting value of 29 pcm/sec bounds the maximum value calculated for the EPU core designs documented in LR Section 2.8.2. The maximum value is calculated using the maximum possible rod speed and maximum differential rod worth for the EPU core. The value of 29 pcm/sec will be confirmed every reload cycle by confirming that the maximum differential rod worth is less than or equal to 38.7 pcm/inch, which is the value corresponding to a reactivity insertion rate of 29 pcm/sec.

SRXB-1.3.27 Identify any limiting safety system settings or limiting conditions for operation that are established or confirmed based upon the safety analysis performed for this section of the licensing report.

This analysis forms the basis for the following safety system settings or limiting condition for operation (LCO):

1. Overtemperature Delta-T (OTΔT) reactor trip setpoints
2. Power range high neutron flux reactor trip setpoint
3. Pressurizer safety valve set pressures
4. Main steam safety valve set pressures
5. Moderator temperature coefficient Technical Specification LCO
6. Rod drop time Technical Specification LCO
7. Axial flux difference Technical Specification LCO
8. Control rod insertion limits Technical Specification LCO
9. Nuclear enthalpy rise hot channel factor Technical Specification LCO
10. Departure from Nucleate Boiling (DNB) parameters Technical Specification LCO

SRXB-1.3.28 Provide graphs correlating the analyzed power level to the maximum reactivity insertion rates. Include the maximum achievable rates based on a reference core design and the maximum permissible rates determined via safety analysis.

Per discussions with the NRC staff during the public meeting held on June 23, 2011, the information being requested for this RAI is the amount of margin between the maximum reactivity insertion rate of 29 pcm/sec (38.7 pcm/inch differential rod worth) assumed in the STDP Rod Withdrawal at Power (RWAP) analysis and the actual values calculated for the EPU core.

The maximum value calculated for the EPU core is [ ]^{a,c} pcm/inch, providing approximately [ ]^{a,c} margin to the analysis value of 38.7 pcm/inch. The maximum differential rod worth will be reconfirmed to be less than 38.7 pcm/inch during the reload design process for the first EPU core and for every subsequent reload core.

<u>**2.8.5.6.1 Inadvertent Opening of Pressurizer Pressure Relief Valve**</u>

SRXB-1.3.29 It is stated, in Section 2.8.5.6.1.3, that "FPL [Florida Power and Light]... concludes that the plant will continue to meet... the requirements of PTN [Turkey Point Nuclear] GDCs [general design criteria] 6 and 29 following the implementation of the proposed EPU [extended power uprate]." Yet, the Technical Evaluation states that the inadvertent opening of a PORV [power operated relief valve] is bounded by the PTN small break loss of coolant accident (SBLOCA) analysis. GDCs 6 and 29 pertain to fuel damage limits and post-anticipated operational occurrence subcriticality. The SBLOCA analysis does not demonstrate compliance with these design criteria, nor is it intended to do so. Please provide a technical evaluation that demonstrates

satisfaction of the referenced GDCs at EPU conditions for the subject event and substantiates the conclusion referenced above.

The Turkey Point units were licensed before the Standard Review Plan (SRP) was issued. As such, an Inadvertent Opening of a Pressurizer Safety Valve (i.e., RCS Depressurization) analysis is not part of the current licensing basis and no analysis was done in support of the Extended Power Uprate. For those plants for which an explicit analysis of this event is performed, it is typically shown that the consequences of the event are not very limiting with respect to DNBR. The transient results in a gradual RCS depressurization accompanied by a slight decrease in DNBR. Therefore, for Turkey Point, it would be expected that the event would be terminated either by the DNB-protective over-temperature ΔT trip or the low pressurizer pressure trip and the DNBR transient would be rather benign. Since RS-001 specifies that it is not the staff's intent to impose new licensing basis requirements on licensees requesting EPUs, it was determined that no explicit analysis was necessary.

2.8.5.6.3.2 Large Break LOCA

SRXB-1.3.30 Demonstrate the validity of the sampling functions for safety injection accumulator cover pressure and safety injection temperature by providing a plot of the sampled probability density function with a histogram of observed pressures and temperatures collected from recent surveillance data.

The accumulator pressure and safety injection (SI) temperature are sampled from their respective distribution for each of the WCOBRA/TRAC calculations. For both accumulator pressure and SI temperature, [

]^{a,c}

Histograms of the sampled values are provided in Figures 1.3.30-1a and 1.3.30-1b for accumulator pressure and SI temperature, respectively. The accumulator pressure range used in the ASTRUM analysis is: 589.7 psia to 714.7 psia. This accumulator pressure range is based on Technical Specification range of 614.7 psia to 689.7 psia with +25 psia uncertainty. The SI temperature range used in the ASTRUM analysis is: 34°F to 105°F. This SI temperature range is based on Technical Specification range of 39 °F to 100°F with ±5°F uncertainty. Histograms of the plant surveillance data are provided in Figures 1.3.30-2a and 1.3.30-2b for accumulator pressure and ambient air (a conservative representation of SI temperature) temperature, respectively. By procedure ambient air temperature is used for the routine surveillance of RWST temperature for convenience since there are no installed temperature elements in the RWST. Actual RWST water temperature would not experience as much variation due to the large heat capacity of the RWST tank inventory. From Figures 1.3.30-2a and 1.3.30-2b, it can be observed that the sampled ranges bound the plant surveillance data, except for one ambient air temperature data point at 26°F. This single data point is not representative of the RWST temperature since, due to the large heat capacity of the RWST, there was insufficient time at this low temperature to lower the RWST water temperature beyond the Technical

Specification range. Therefore, it is considered appropriate to not include this data point in the SI temperature sampling range. Since the sampled accumulator pressure and SI temperature ranges bound both the Technical Specification limits and the plant surveillance data (except for one ambient air temperature as discussed above) and a conservative uniform distribution is assumed, the accumulator pressure and SI temperature ranges sampled in the ASTRUM analysis are considered acceptable.

a,c

Figure 1.3.30-1a: Sampled Accumulator Pressure

a,c

Figure 1.3.30-1b: Sampled Safety Injection Temperature

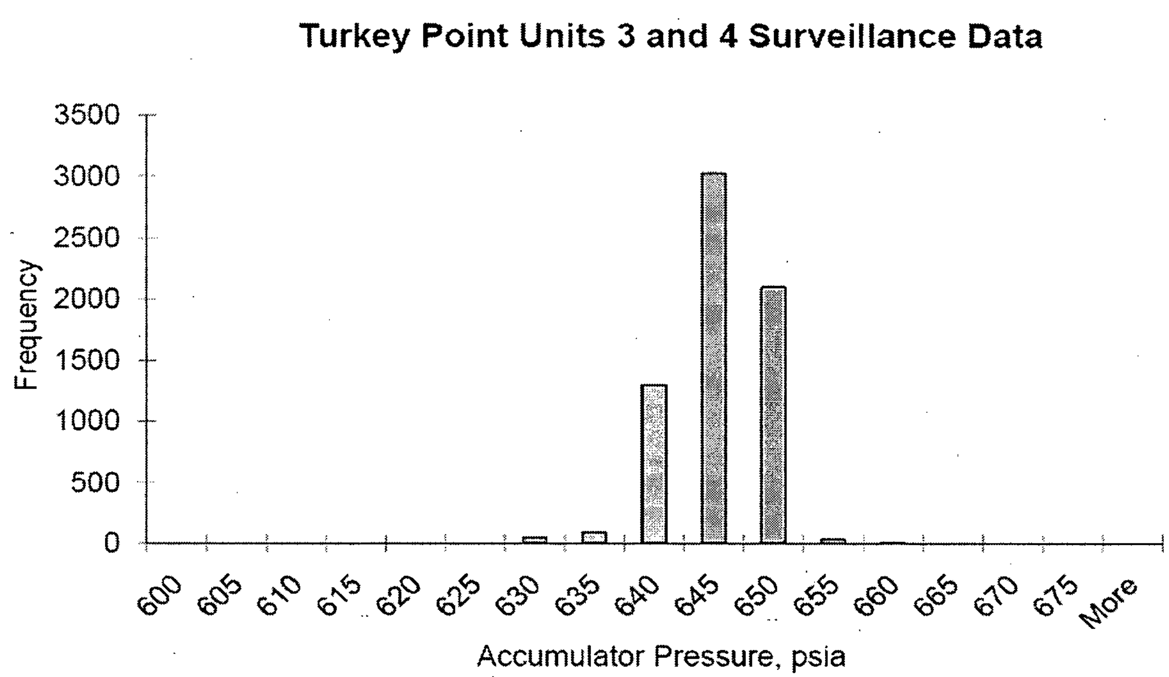


Figure 1.3.30-2a: Plant Surveillance Accumulator Pressure

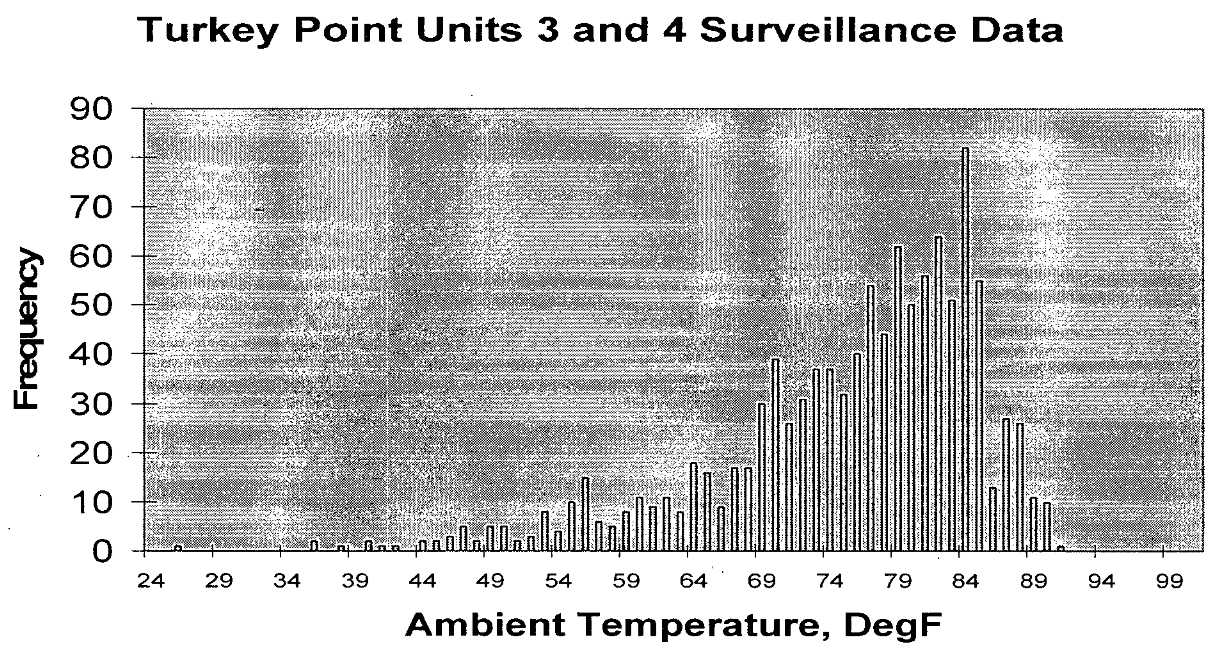


Figure 1.3.30-2b: Plant Surveillance Ambient (SI) Temperature

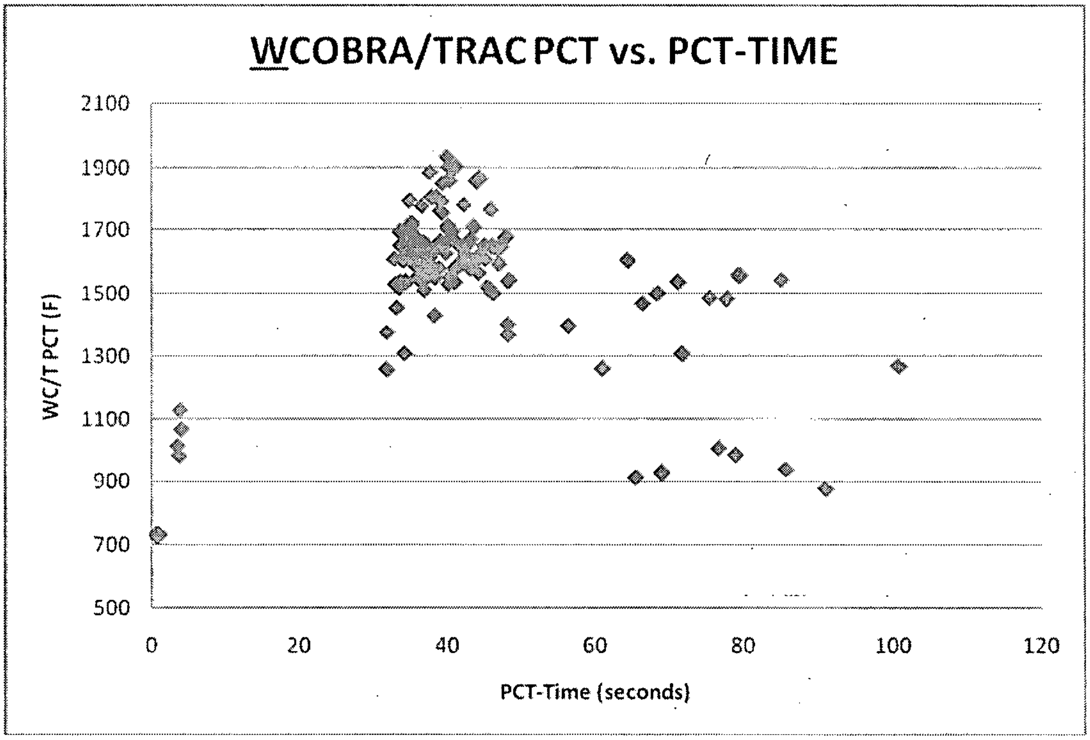
SRXB-1.3.31 Provide the basis for the analyzed single failure assumption. If the basis is generic, provide a Turkey Point specific justification for use of the generic assumption.

The analysis is performed to demonstrate that the ECCS would provide abundant emergency core cooling to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant and satisfy the criteria of 10 CFR 50.46. The most limiting single active failure is determined to assure that the failure mode selected provides the maximum impairment of the ECCS safety function to provide adequate core cooling. The most limiting single active failure has been generically determined (WCAP-16009-P-A, Section 12-3-4) to be the loss of a train of ECCS. This assumption maximizes the reduction in available ECCS flow to the core. This results in the loss of a LHSI and HHSI pump flow to the core.

Turkey Point has a HHSI configuration that automatically upon a safety injection signal aligns the two HHSI pumps from the unaffected unit to provide flow to the core of the affected unit. This is not a change in the design configuration of the Turkey Point units. The unaffected unit has two trains of ECCS independent from the two trains of ECCS of the affected unit. The emergency operating procedures instruct the operator to assure that two HHSI pumps from either unit remain operating to deliver flow to the core of the affected unit. In the most limiting single active failure scenario of the loss of a train of ECCS the operator secures (shuts off) one HHSI pump from the unaffected unit while maintaining flow from one HHSI pump from the unaffected unit and one HHSI pump from the affected unit. Therefore, the minimum ECCS flows used in the analysis were based on flows supplied from one LHSI pump and two HHSI pumps.

SRXB-1.3.32 Provide a scatter plot of PCT [peak cladding temperature] vs. time-of-PCT for the population of analyzed cases.

The WCOBRA/TRAC PCT versus PCT-time for all 124 ASTRUM cases is provided in Figure 1.3.32-1 below. Please note that the WCOBRA/TRAC PCT and PCT-time data are also provided in Table 1.3.34-2 in response to RAI SRXB-1.3.34..



- **Figure 1.3.32-1: WCOBRA/TRAC PCT versus PCT-Time**

SRXB-1.3.33 For the most severe analyzed case, provide the assumed fuel burnup.

The limiting Peak Cladding Temperature (PCT)/Maximum Local Oxidation (MLO) case corresponding to LR Tables 2.8.5.6.3.2-2 and 2.8.5.6.3.2-3 and LR Figure 2.8.5.6.3.2-2 resulted from a Hot Rod with a burnup of 1553 MWD/MTU, and the limiting Core-Wide Oxidation (CWO) case corresponding to LR Table 2.8.5.6.3.2-2 and LR Figure 2.8.5.6.3.2-3 resulted from a Hot Rod with a burnup of 834 MWD/MTU. It is noted that limiting cases were Hot Rods with burnup values near beginning-of-life (BOL).

SRXB-1.3.34 Tabulate the initial conditions, operating parameters, PCT, time of PCT, accumulator empty time, and safety injection initiation time for the following cases (with respect to PCT):

-3 highest
-Upper quartile
-Median, immediately higher, and immediately lower
-Lower quartile
-3 lowest

A data file is acceptable (and preferred).

The initial conditions and operating parameters (specifically, the sampled input parameters listed in LR Table 2.8.5.6.3.2-1) are provided in Table 1.3.34-1. The other parameters (steam generator tube plugging, reactor power, safety injection flow, etc.) from LR Table 2.8.5.6.3.2-1 not included in Table 1.3.34-1 herein are not sampled in the uncertainty analysis and are set at their bounding values. The WCOBRA/ TRAC (WC/T) PCT, PCT-time, Safety Injection (SI) initiation time and accumulator empty time are provided in Table 1.3.34-2. Note that the data is provided for all 124 ASTRUM cases, ranked by HOTSPOT PCT (from highest to lowest). In addition a compact disk (CD) containing the data in both Tables 1.3.34-1 and 1.3.34-2 is provided as requested.

a,c

Table 1.3.34-1: Turkey Point Units 3 and 4 ASTRUM Uncertainty Attributes

a,c

Table 1.3.34-1: Turkey Point Units 3 and 4 ASTRUM Uncertainty Attributes

a,c

Table 1.3.34-1: Turkey Point Units 3 and 4 ASTRUM Uncertainty Attributes

a,c

Table 1.3.34-1: Turkey Point Units 3 and 4 ASTRUM Uncertainty Attributes

a,c

Table 1.3.34-1: Turkey Point Units 3 and 4 ASTRUM Uncertainty Attributes

a,c

Table 1.3.34-1: Turkey Point Units 3 and 4 ASTRUM Uncertainty Attributes

a,c

Table 1.3.34-2: WCOBRA/TRAC Run Info

a,c

Table 1.3.34-2: WCOBRA/TRAC Run Info

a,c

Table 1.3.34-2: WCOBRA/TRAC Run Info

a,c

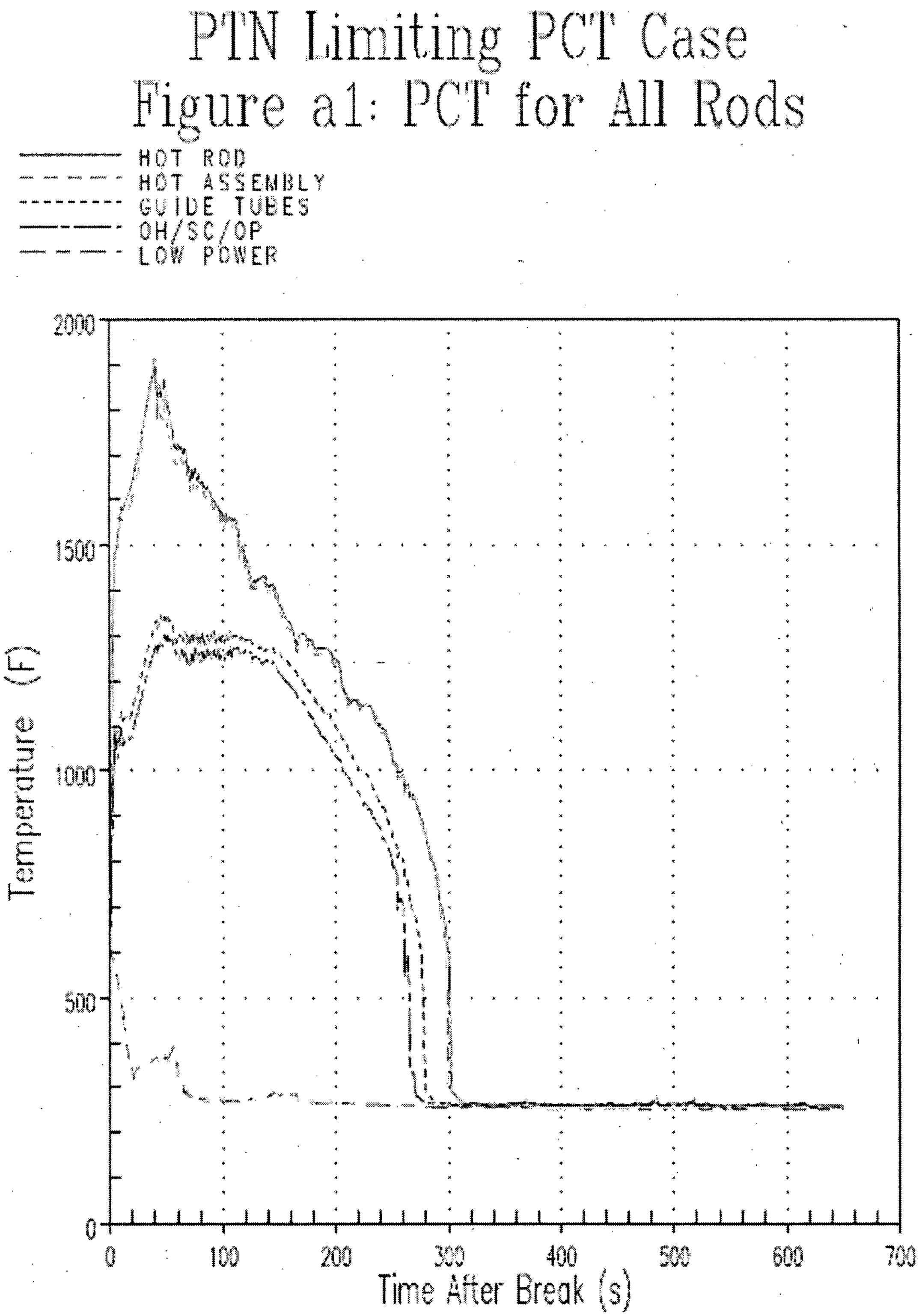
Table 1.3.34-2: WCOBRA/TRAC Run Info

SRXB-1.3.35 Explain the cause of the abrupt drop in low-power channel liquid level observable at approximately 350 seconds in Figure 2.8.5.6.3.2-13.

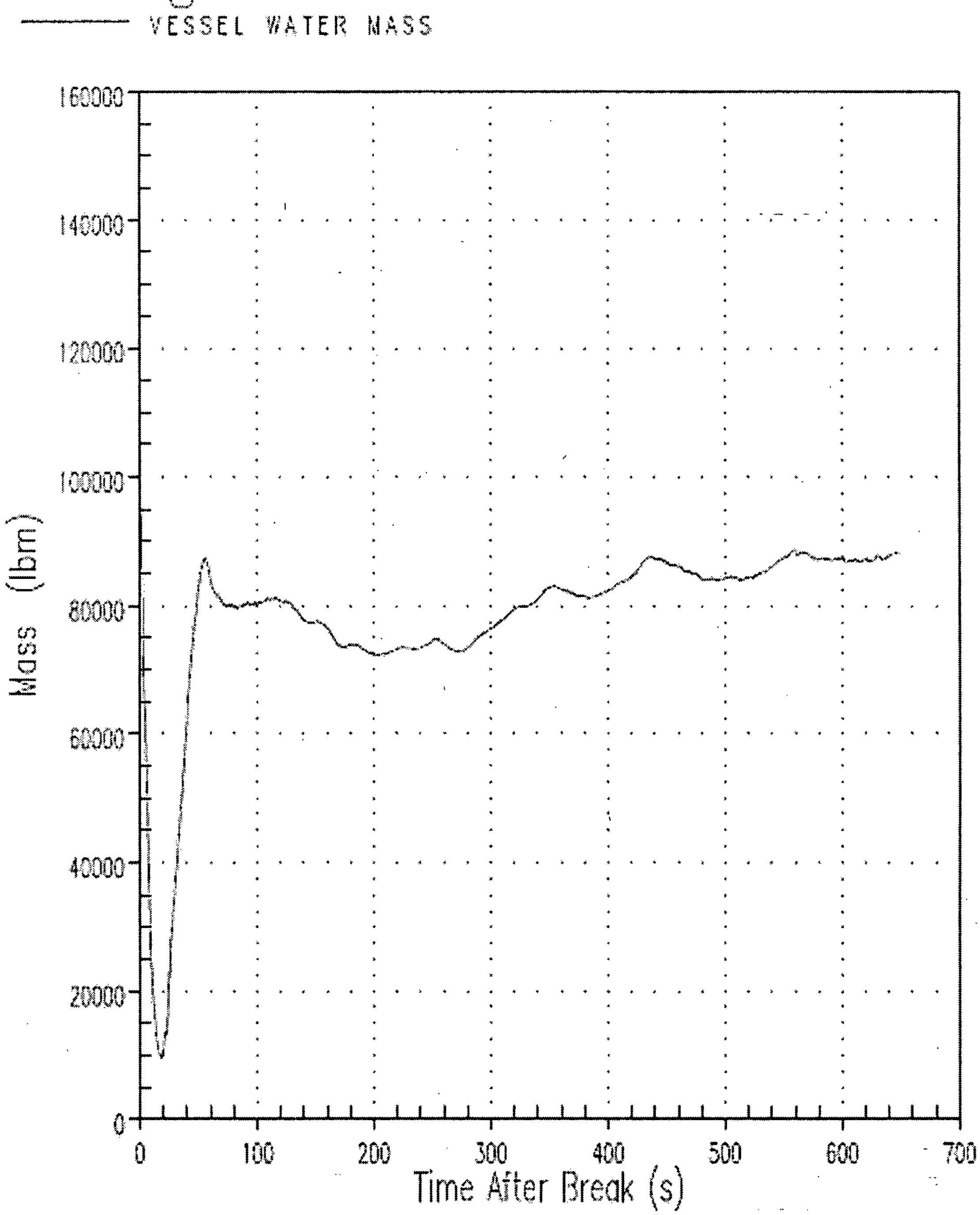
As shown in LR Table 2.8.5.6.3.2-3, Hot Rod PCT is calculated to occur at approximately 40 seconds, and as shown in LR Figure 2.8.5.6.3.2-15, the core is fully quenched at approximately 310 seconds. Therefore, the abrupt drop in low-power channel liquid level observed at approximately 350 seconds is well after the PCT is calculated and all rods have quenched.

To confirm that the core liquid levels continue to increase and all rods remain quenched after 350 seconds, the limiting PCT transient was re-run with the transient time extended another 300 seconds. From Figure a1, it can be observed that all rods remain quenched, and vessel water mass (Figure a2), lower plenum collapsed liquid level (Figure a3) and core liquid levels (Figure a4) keep increasing and are well above that required for the core to remain covered by a two-phase mixture with all fuel rods quenched as boiling continues. In addition, from Figure a5, it can be observed that the liquid level in the downcomer becomes stable after about 450 seconds.

The primary cause of the oscillations (including the abrupt increase and decrease in liquid level observed in the low-power channel) in collapsed liquid levels is an imbalance in the vessel hydrostatics due to redistribution of the liquid inventory in the vessel. During the time period in question (350 seconds and beyond), as the decay heat gradually decreases, core steam flow also decreases which allows increased draining of liquid inventory into the core from the upper plenum. As the liquid enters the core (note that liquid enters the core from the upper plenum is being shown as negative flow) from the upper plenum (flow at the top of the Low Power channel is shown in Figure a6), there is a tendency for cross-flow to take place between the core channels (cross-flow between Low Power channel and adjacent average channel is shown in Figure a7) which redistributes the increasing core inventory. The increasing core liquid inventory and redistribution of the inventory between the core channels further affects the hydrostatic balance in the vessel which leads to the calculated oscillations. Note that the described hydrostatic instability is calculated to occur periodically during the post-quench period. However, as mentioned earlier, the downcomer level remains relatively stable (Figure a5) and the vessel inventory (Figure a2) gradually increases during the post-quench period beyond 300 seconds.

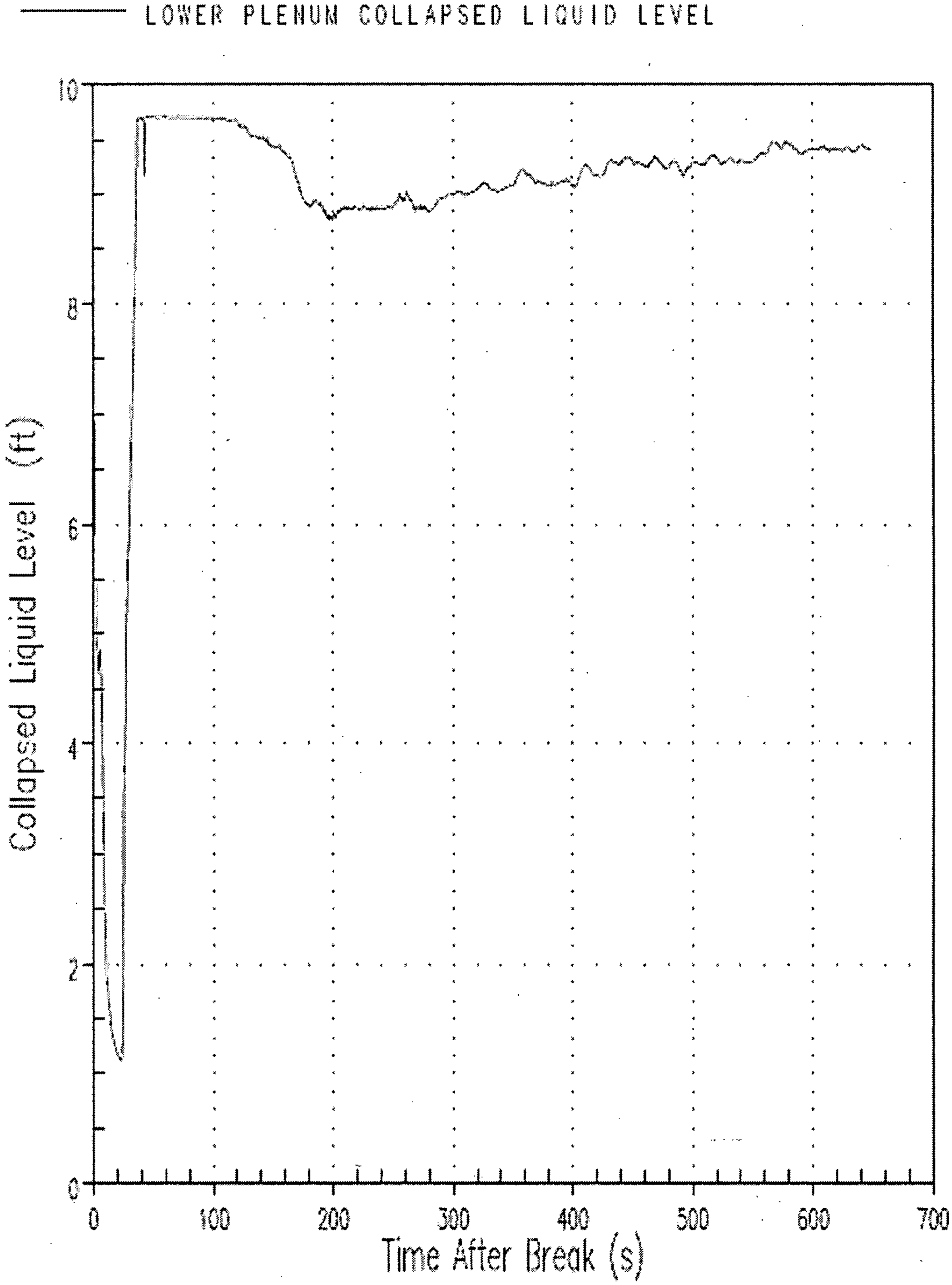

PTN Limiting PCT Case
Figure a1: PCT for All Rods
HOT ROD
HOT ASSEMBLY
GUIDE TUBES
OH/SC/OP
LOW POWER
Temperature (F)
Time After Break (s)
0
500
1000
1500
2000
100
200
300
400
500
600
700

PTN Limiting PCT Case
Figure a2: Vessel Water Mass



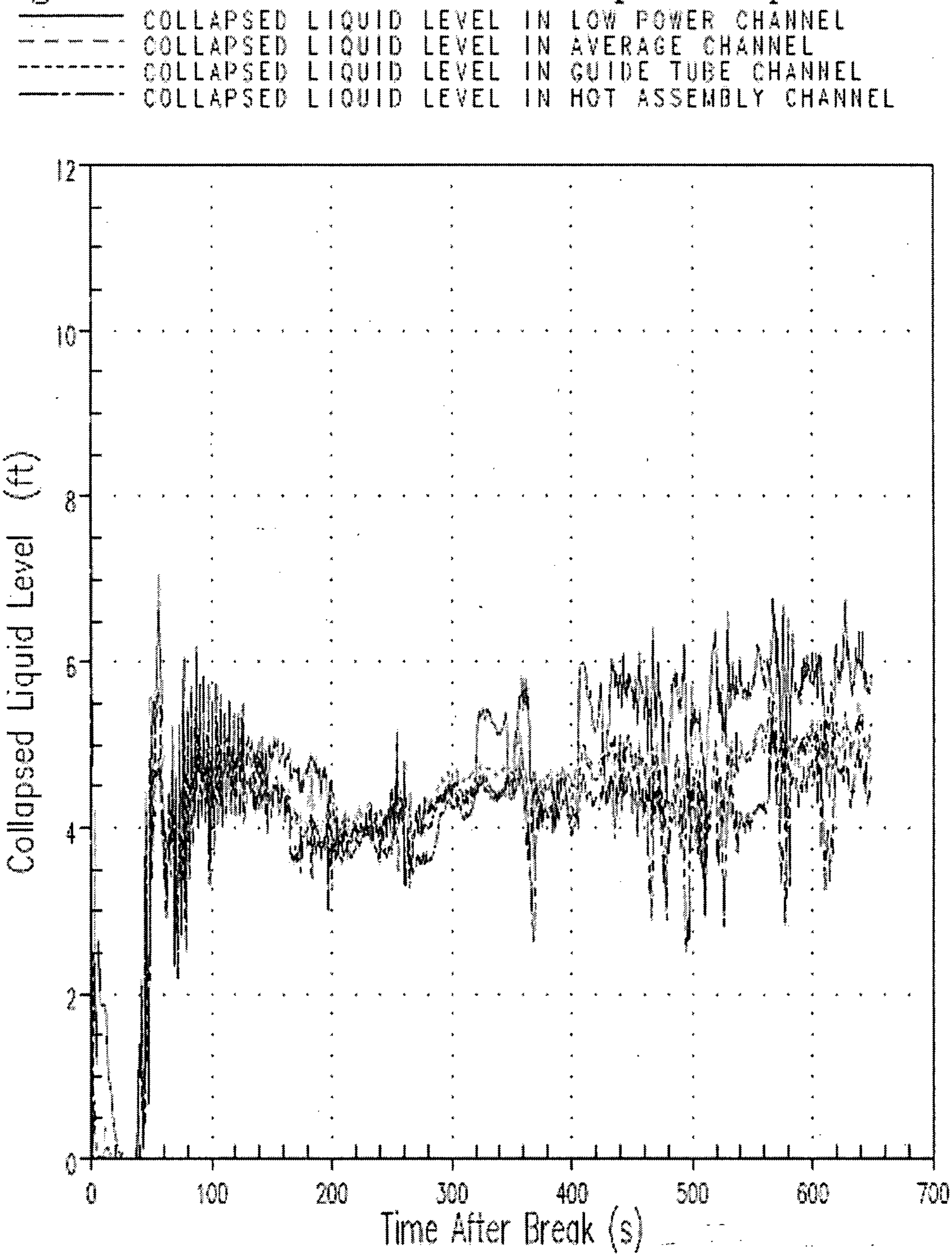
PTN Limiting PCT Case

Figure a3: Lower Plenum Collapsed Liquid Level



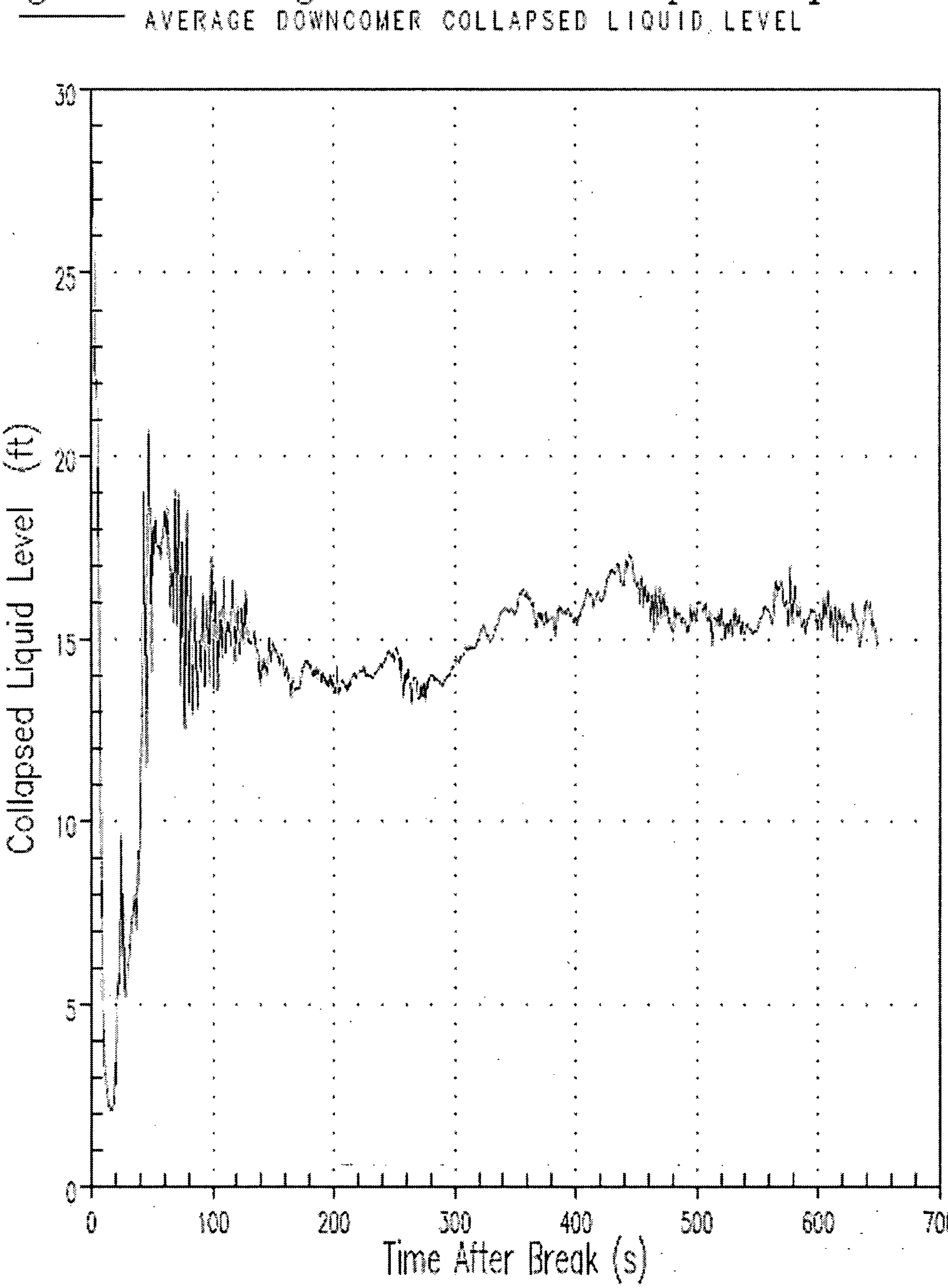
PTN Limiting PCT Case

Figure a4: Core Channels Collapsed Liquid Levels



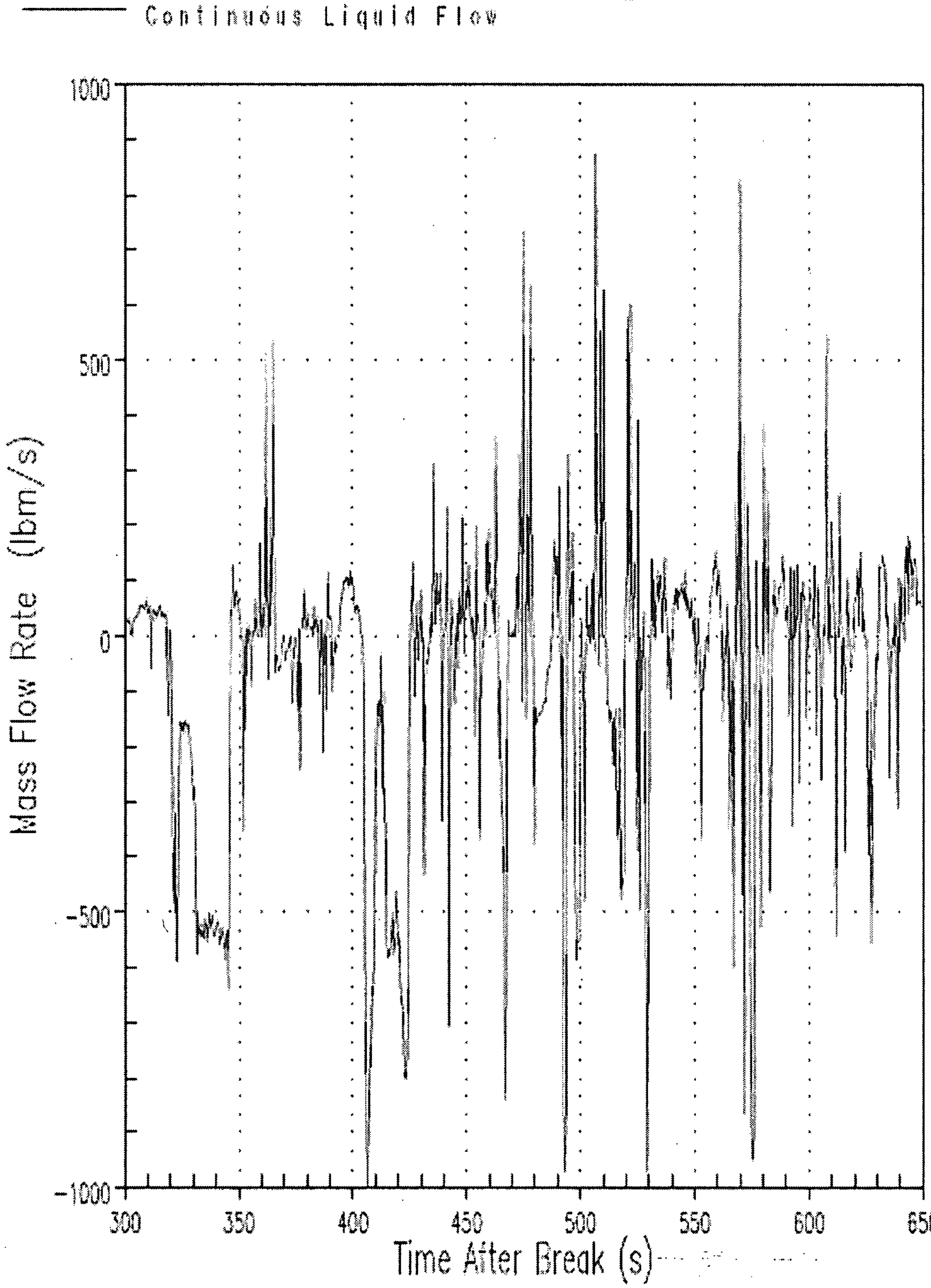
PTN Limiting PCT Case

Figure a5: Average Downcomer Collapsed Liquid Level



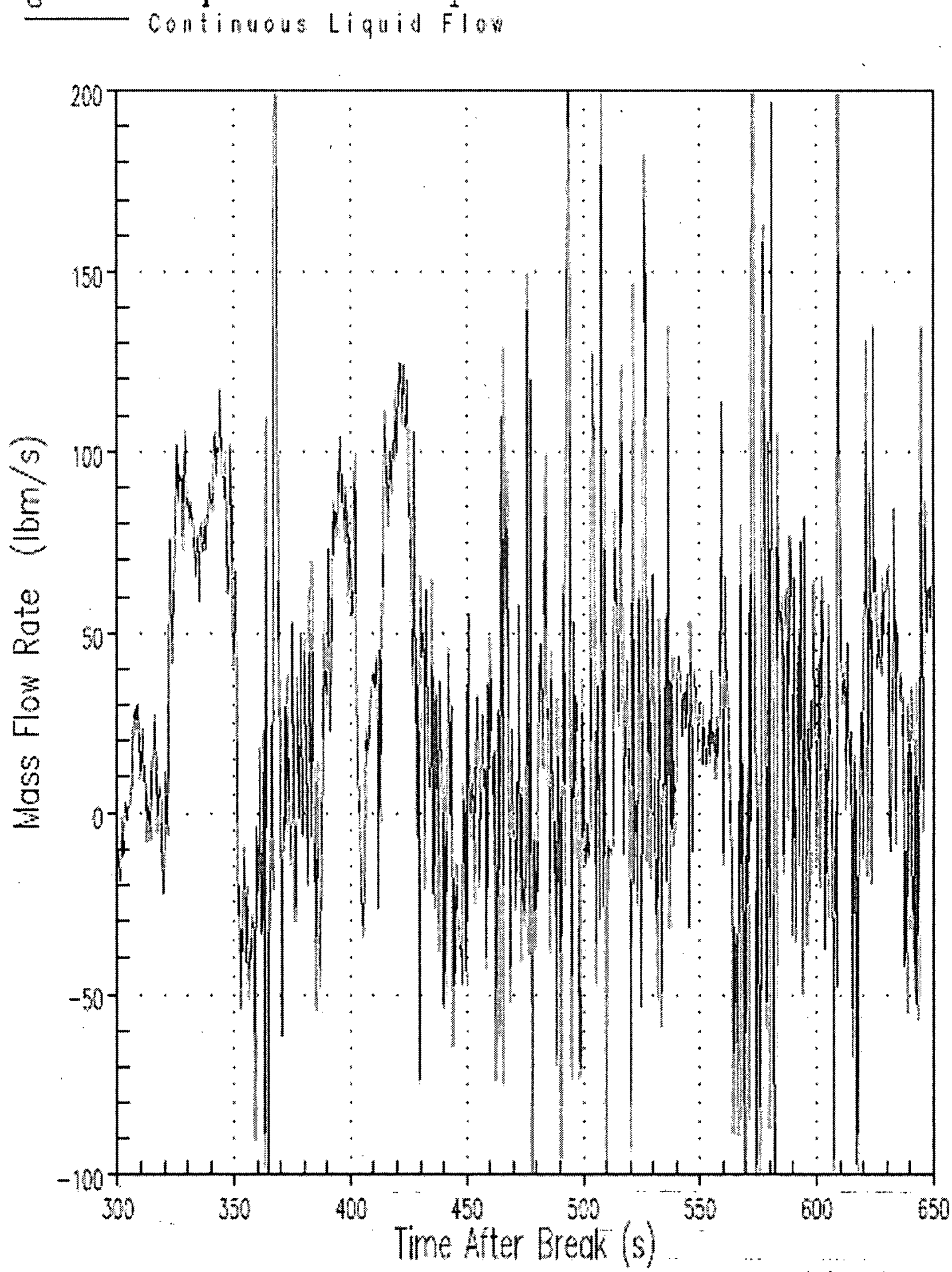
PTN Limiting PCT Case

Figure a6: Continuous Liquid Flow at the Top of Low Power Channel



PTN Limiting PCT Case

Figure a7: Gap Continuous Liquid Flow from Low Power Channel



SRXB-1.3.36 Information provided in Attachment 1 to L-2011-028 describes the Turkey Point LBLOCA nodalization. Provide the following additional clarifications:

a. For gaps interconnecting the downcomer channels, provide the k-factors.

In order to appropriately address this comment, the WCOBRA/TRAC noding diagram for Turkey Point Units 3 and 4 with nine downcomer channel stacks is presented as Figures w1 to w4.

In Figure w1, the numbers enclosed in squares represent channel numbers. Channels are used to define vertical connections in the vessel model. The numbers enclosed in circles represent gap numbers. Gaps are used to define lateral connections in the vessel model. The gap numbers which have an arrow through them connect the channels shown at the start and end of the arrow as shown in Figures w2 to w4.

The downcomer region is modeled with the long vertical channel stacks shown on the outer portion of the noding diagram (Figure w1). Cross-sections of the vessel noding at the different section elevation are presented in Figures w2 to w4. Note that the cold leg and hot leg connection locations can be seen in Figure w3 (Section 6, nozzle region).

Form and wall drag in gap K is specified for the transverse momentum equations using the parameters WKR(K) and FWALL(K). WKR(K) is the form drag loss coefficient (velocity head) **[**

] [a,c]

The single-phase pressure drop, due to wall friction, between two adjacent channels through the gap is then calculated as:

[ ]a,c

[

[ ]a,c

[ ]a,c

[ ]a,c

The total lateral drag coefficient is calculated as:

[ ]a,c

[

]a,c

Refer to Section 4-2 of the CQD (Reference 8) for additional information on WCOBRA/TRAC momentum transfer models.

a,c

Figure w1: Turkey Point Units 3 and 4 WCOBRA/TRAC Vessel Model Noding Diagram

a,c

Figure w2: Turkey Point Units 3 and 4 WCOBRA/TRAC Vessel Sections 1 to 3 (Horizontal View)

a,c

Figure w3: Turkey Point Units 3 and 4 WCOBRA/TRAC Vessel Sections 4 to 6 (Horizontal View)

a,c

Figure w4: Turkey Point Units 3 and 4 WCOBRA/TRAC Vessel Sections 7 to 9 (Horizontal View)

b. Provide plots of nodal void fraction vs. time and azimuthal and axial nodal mass flows vs. time for the following axial channels and azimuthal sections:

i. Axial: Channels 79, 32, 76, 77 (broken loop hot leg and cold leg channels, and channels roughly opposite), and channels below.

ii. Azimuthal: Sections 2-6 (lower plenum, core, CCFL region, upper plenum below nozzles, and nozzle region).

Plots of void fraction vs. time and azimuthal and axial total mass flows vs. time for all downcomer channels from Sections 2 to 6 are provided on following pages.

- Plots of void fraction versus time are shown as Figures b1 to b72.

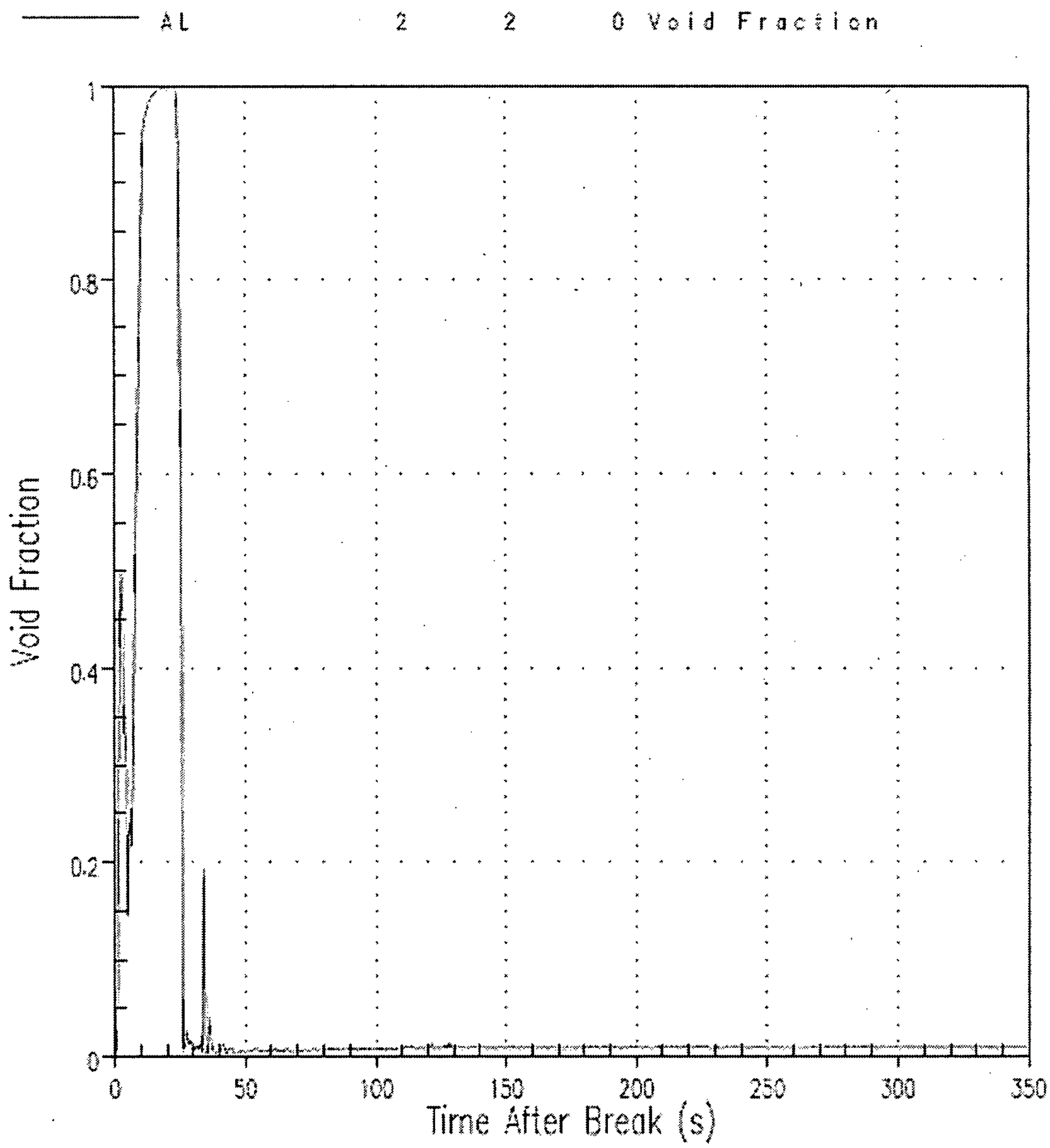
- Plots of total axial mass flow (liquid + vapor + entrained) versus time are shown as Figures c1 to c72.

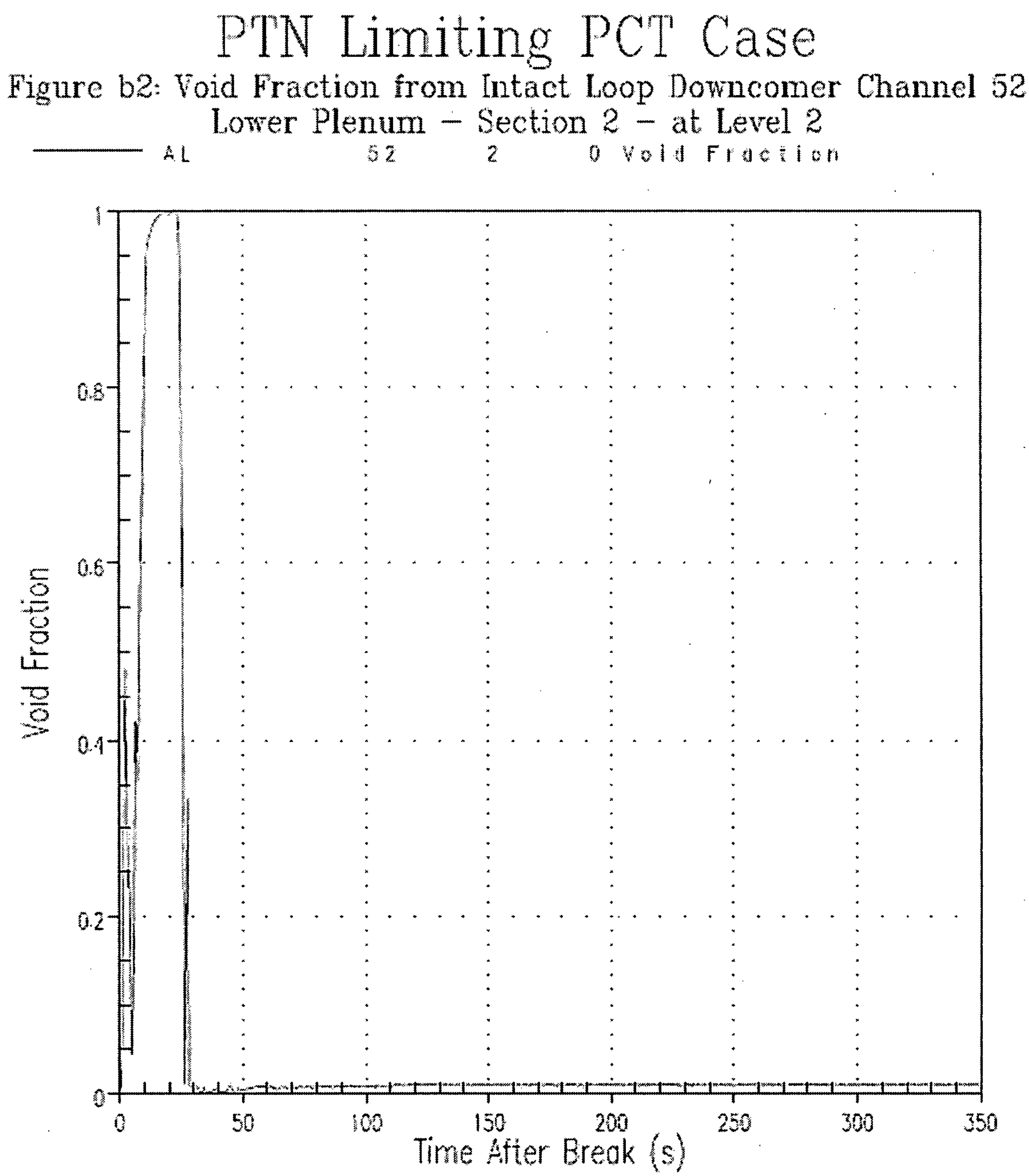
- Plots of total cross-flow (liquid + vapor + entrained from gap) versus time are shown as Figures d1 to d78. Note that Gaps 85, 87 and 89 in nozzle Section 6 are blocked due to the hot leg nozzle penetrations and are not included, since the flow is zero across these gaps.

Note that a compact disk (CD) containing the void fraction versus time from Figures b1 to b72, total mass flow versus time from Figure c1 to c72 and total cross-flow from Figures d1 to d78 is provided as requested.

PTN Limiting PCT Case

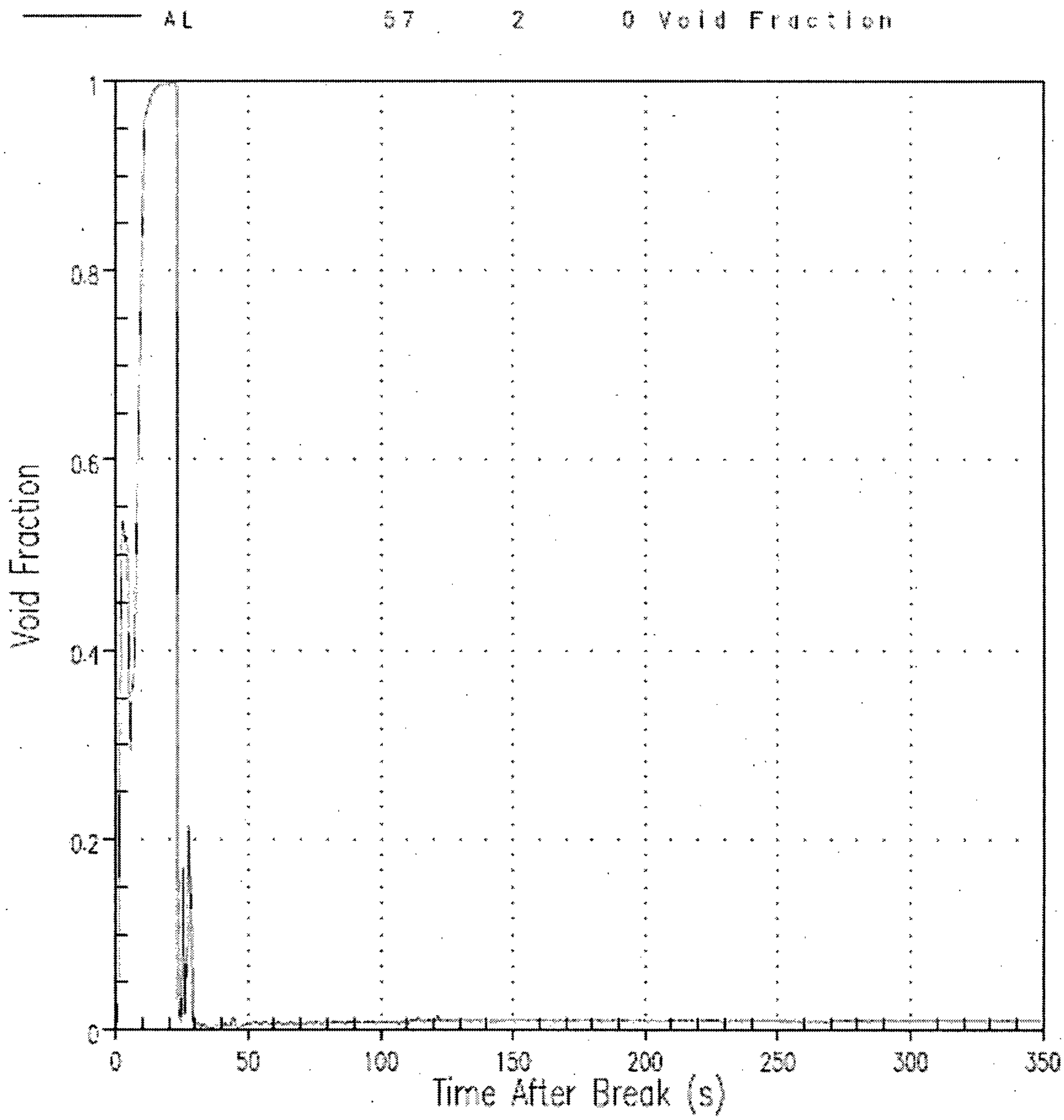
Figure b1: Void Fraction from Intact Loop Downcomer Channel 2
Lower Plenum - Section 2 - at Level 2




PTN Limiting PCT Case
Figure b2: Void Fraction from Intact Loop Downcomer Channel 52
Lower Plenum - Section 2 - at Level 2
AL 52 2 0 Void Fraction
Void Fraction
1
0.8
0.6
0.4
0.2
0
0
50
100
150
200
250
300
350
Time After Break (s)

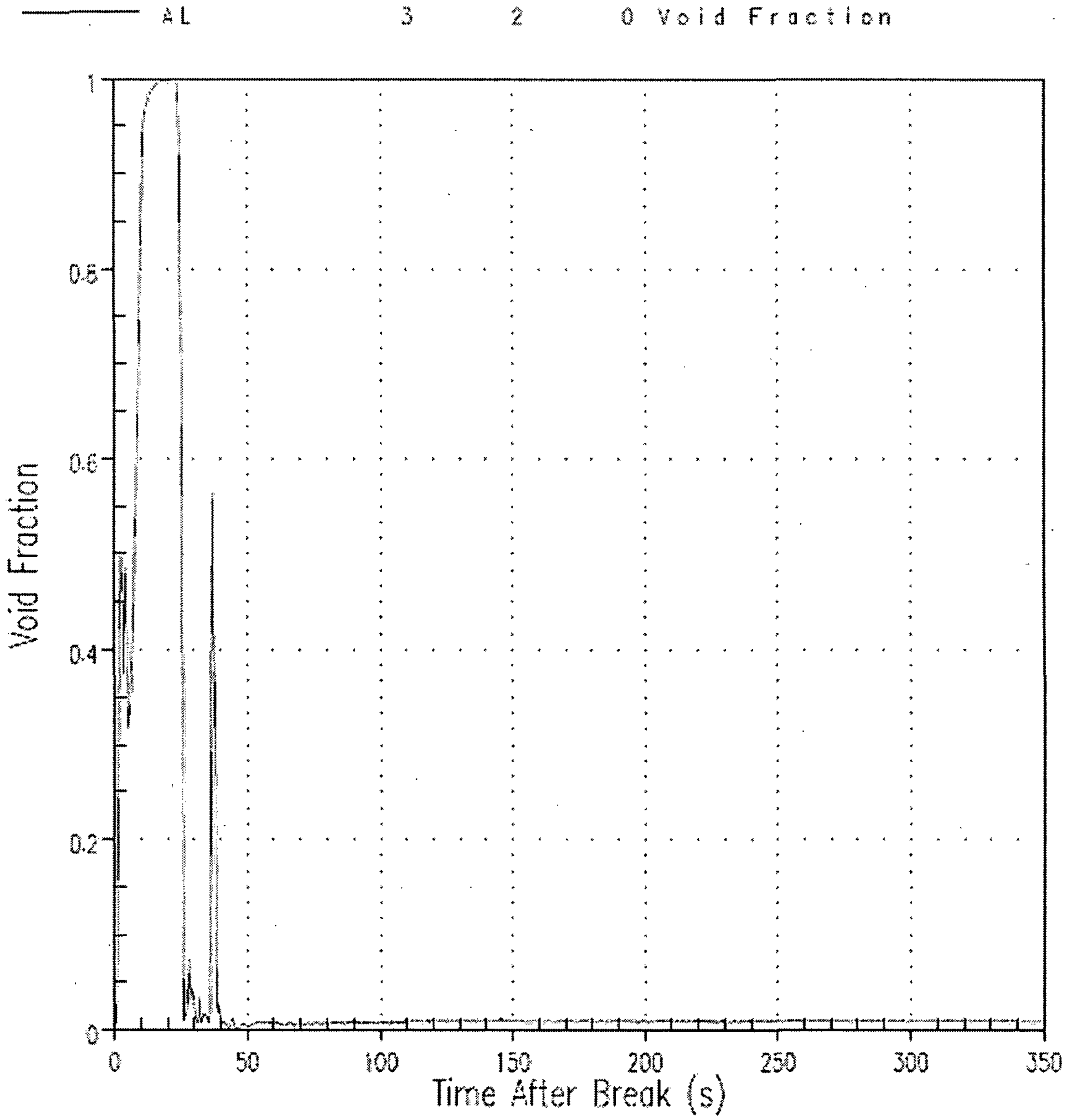
PTN Limiting PCT Case

Figure b3: Void Fraction from Intact Loop Downcomer Channel 57
Lower Plenum – Section 2 – at Level 2



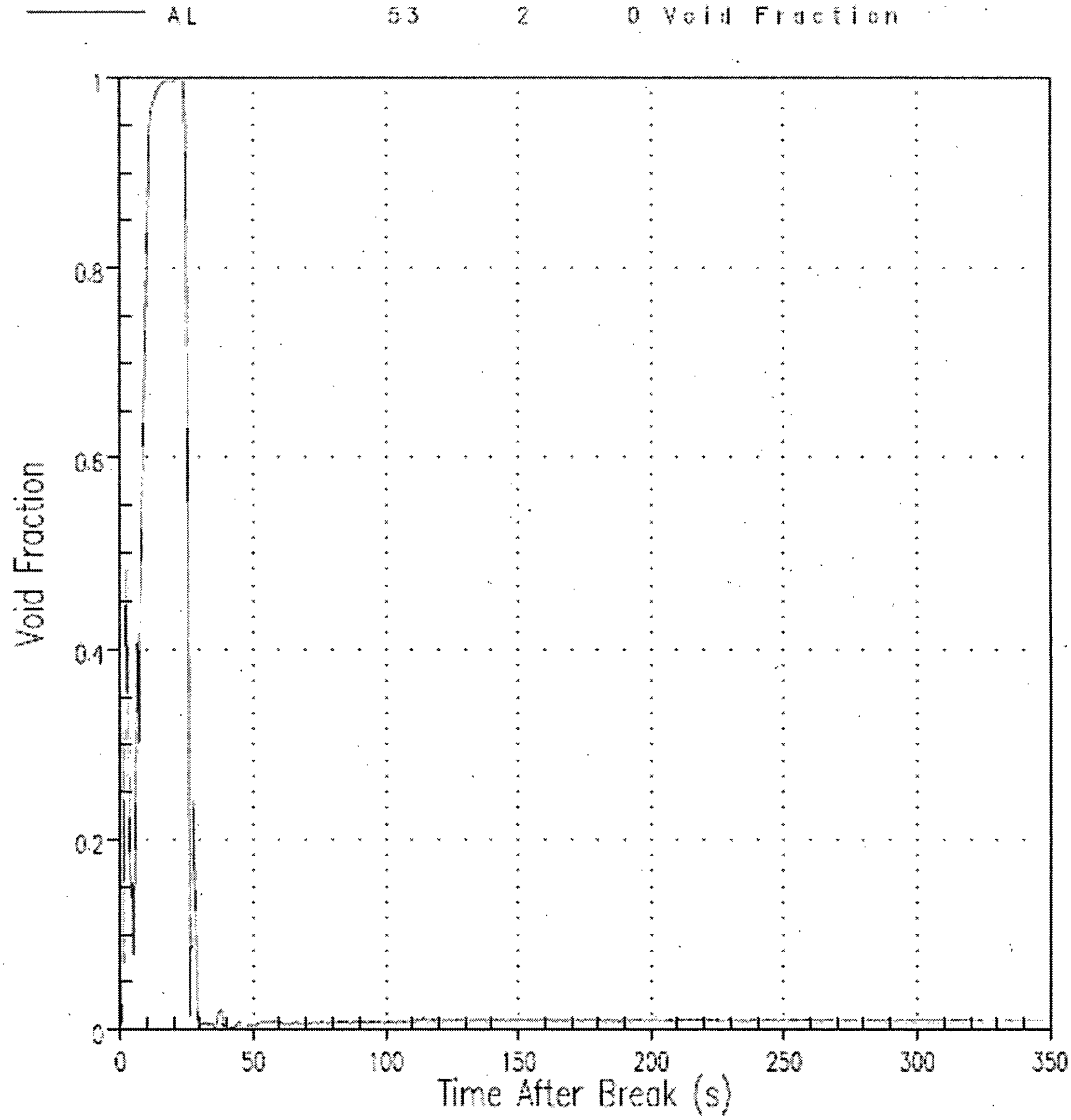
PTN Limiting PCT Case

Figure b4: Void Fraction from Intact Loop Downcomer Channel 3 Lower Plenum – Section 2 – at Level 2



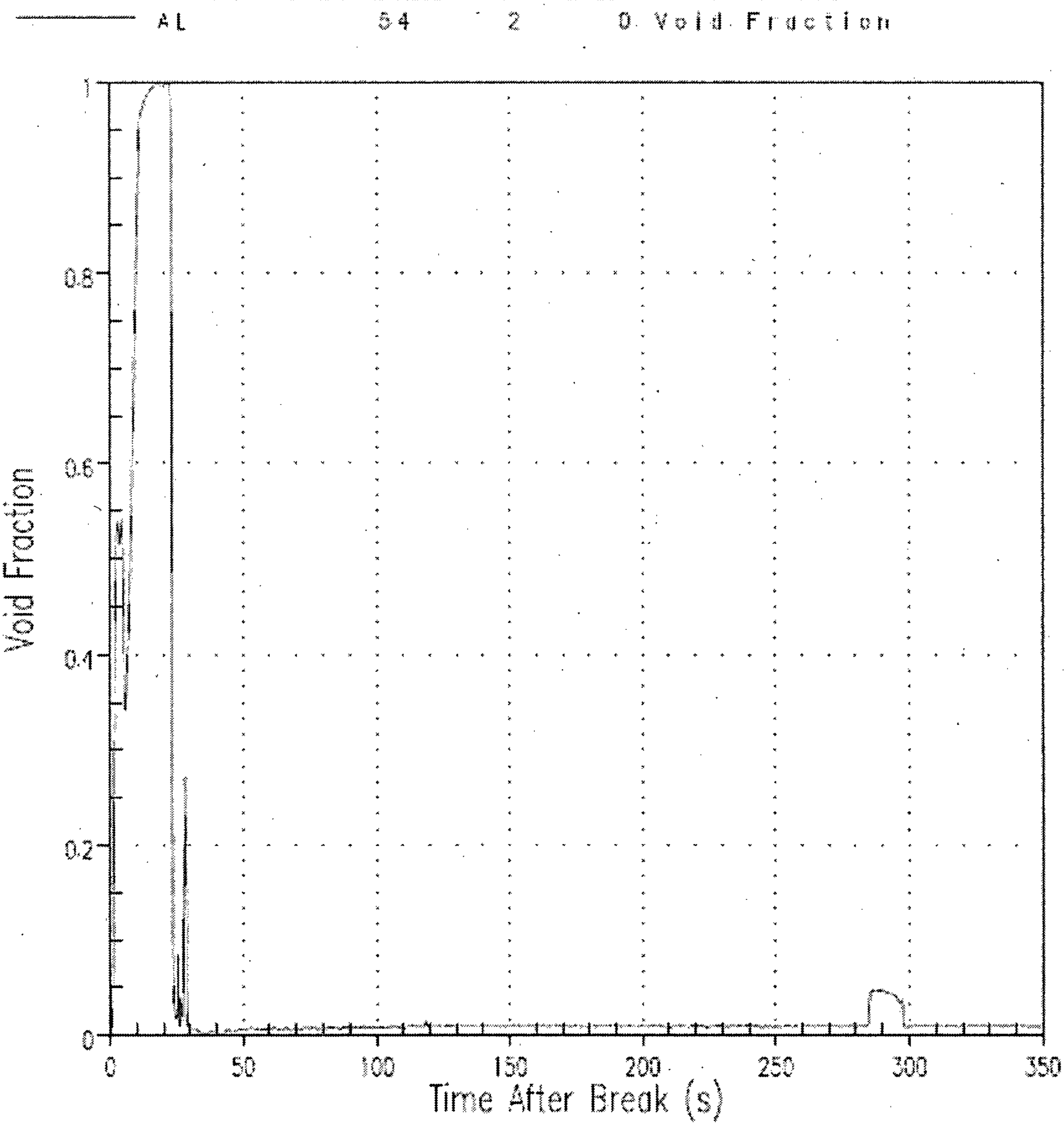
PTN Limiting PCT Case

Figure b5: Void Fraction from Intact Loop Downcomer Channel 53
Lower Plenum – Section 2 – at Level 2



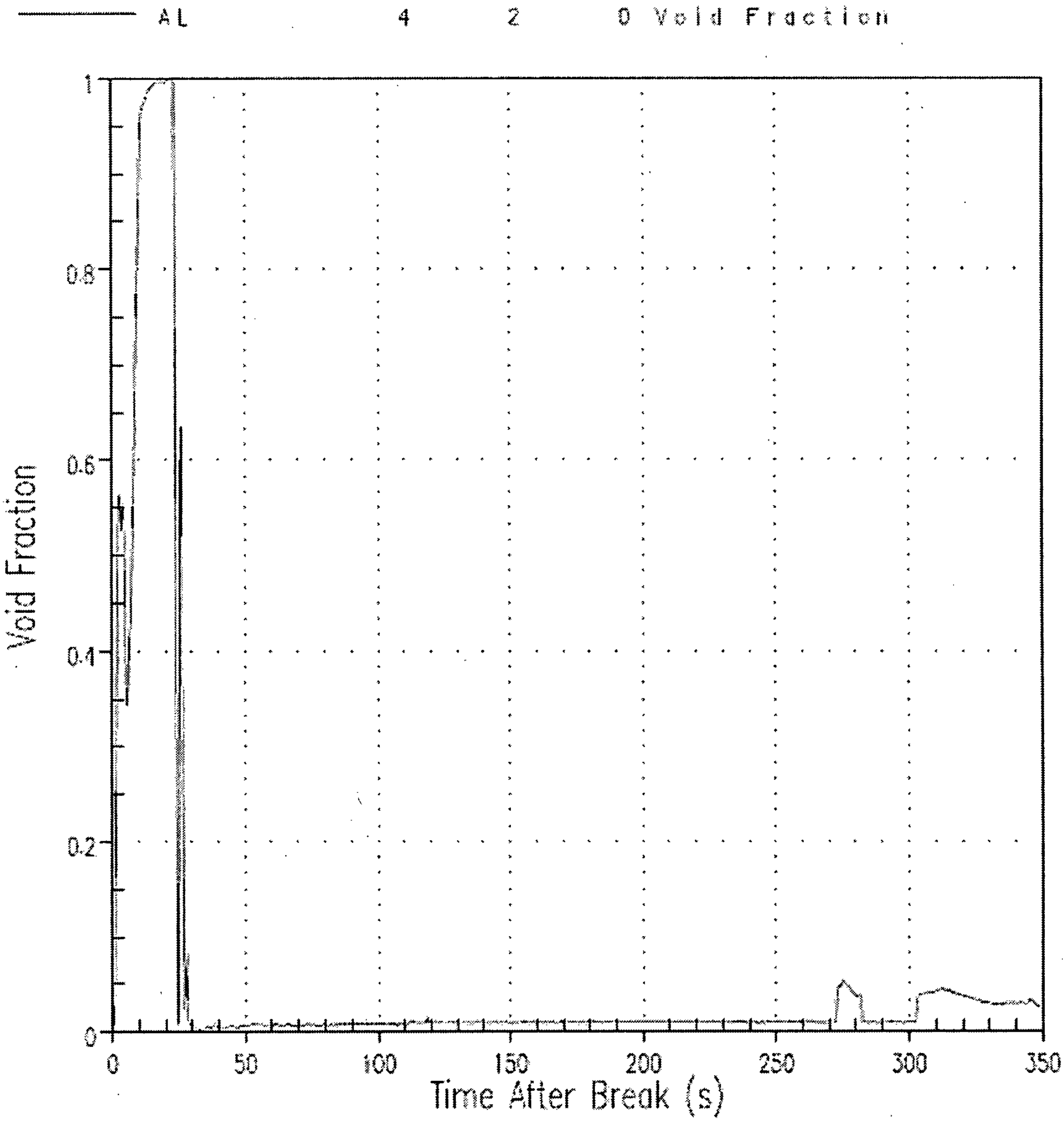
PTN Limiting PCT Case

Figure b6: Void Fraction from Intact Loop Downcomer Channel 54
Lower Plenum – Section 2 – at Level 2



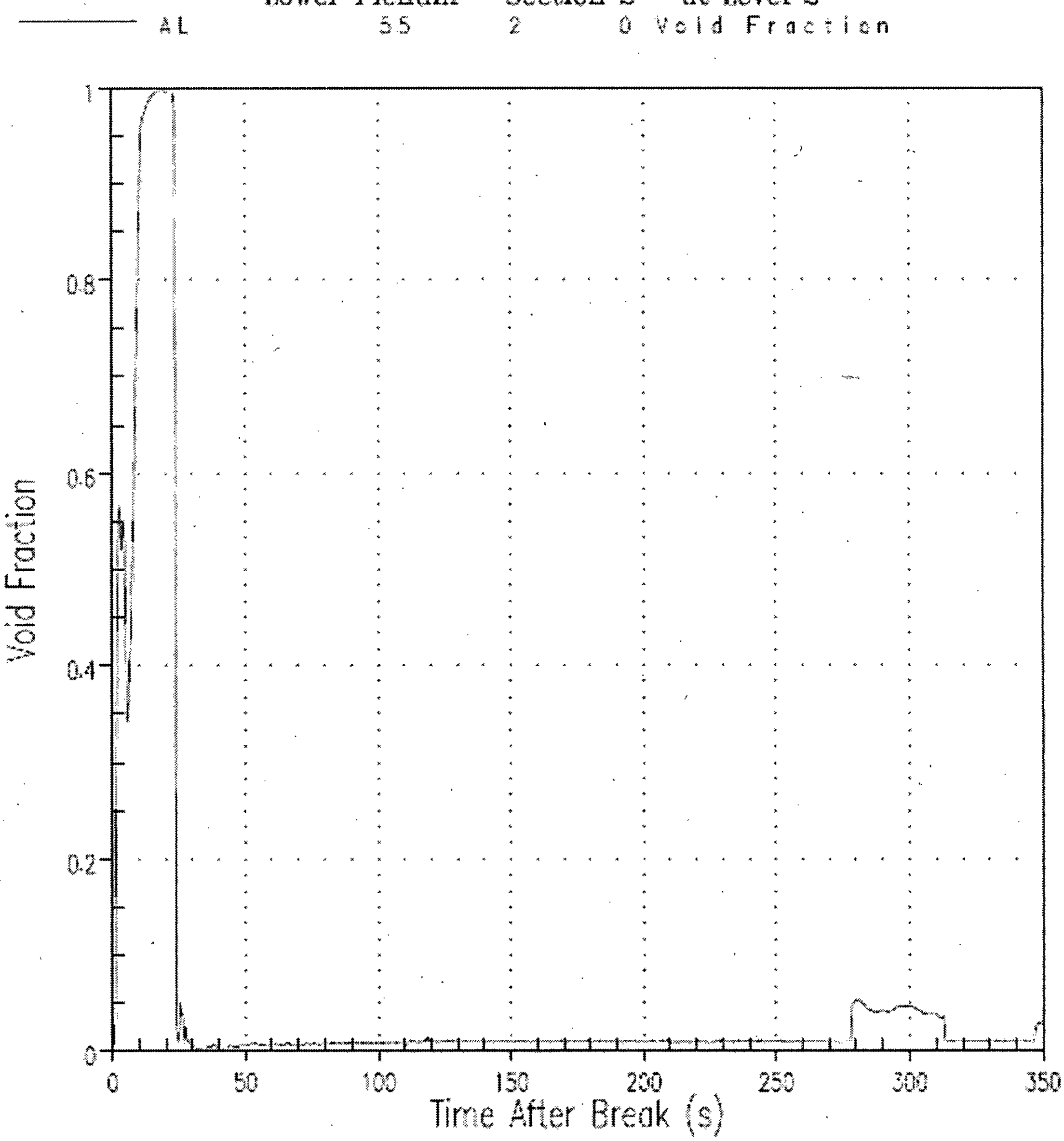
PTN Limiting PCT Case

Figure b7: Void Fraction from Broken Loop Downcomer Channel 4
Lower Plenum – Section 2 – at Level 2



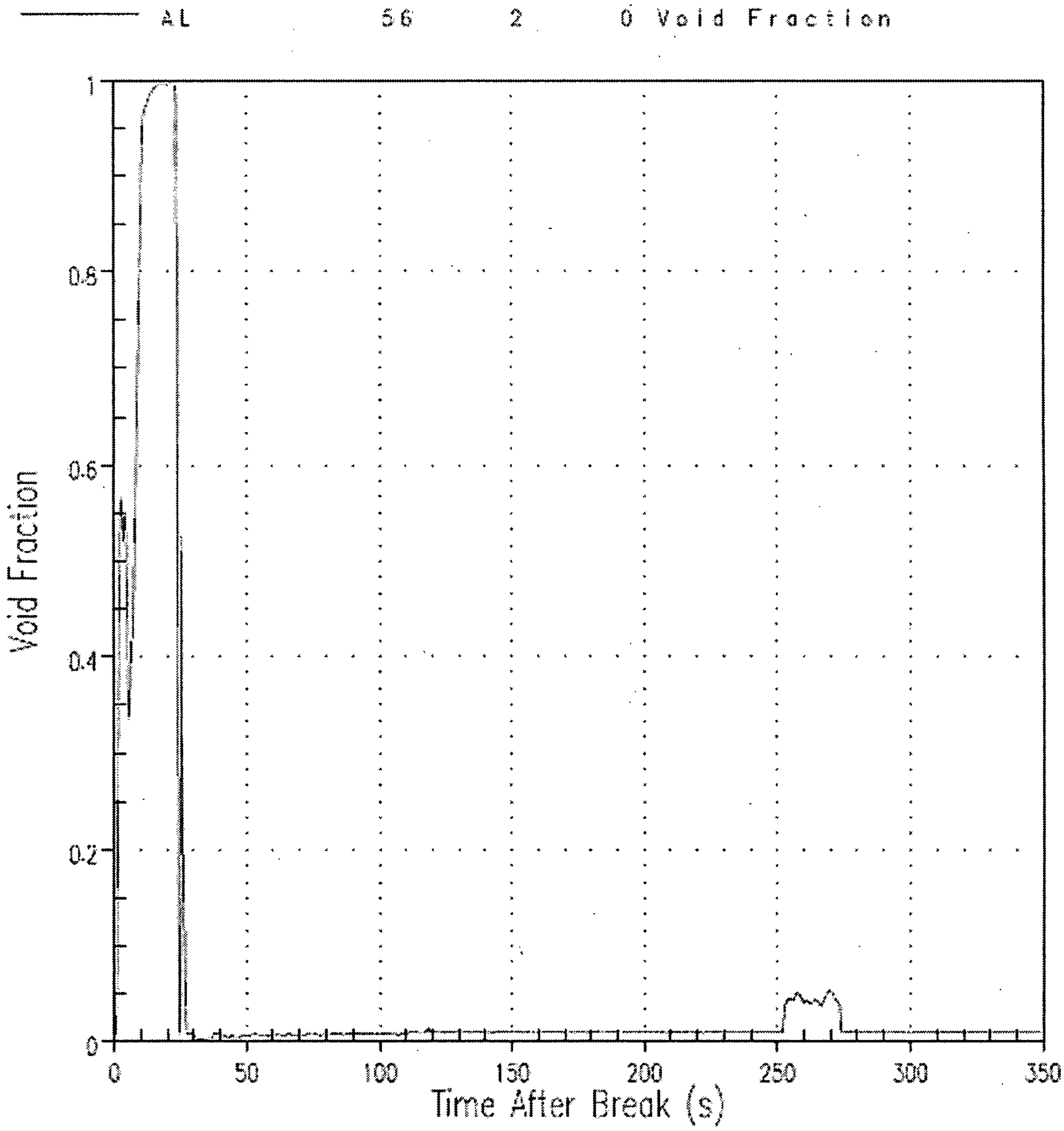
PTN Limiting PCT Case

Figure b8: Void Fraction from Adjacen to Broken Loop Downcomer Channel 55
Lower Plenum – Section 2 – at Level 2



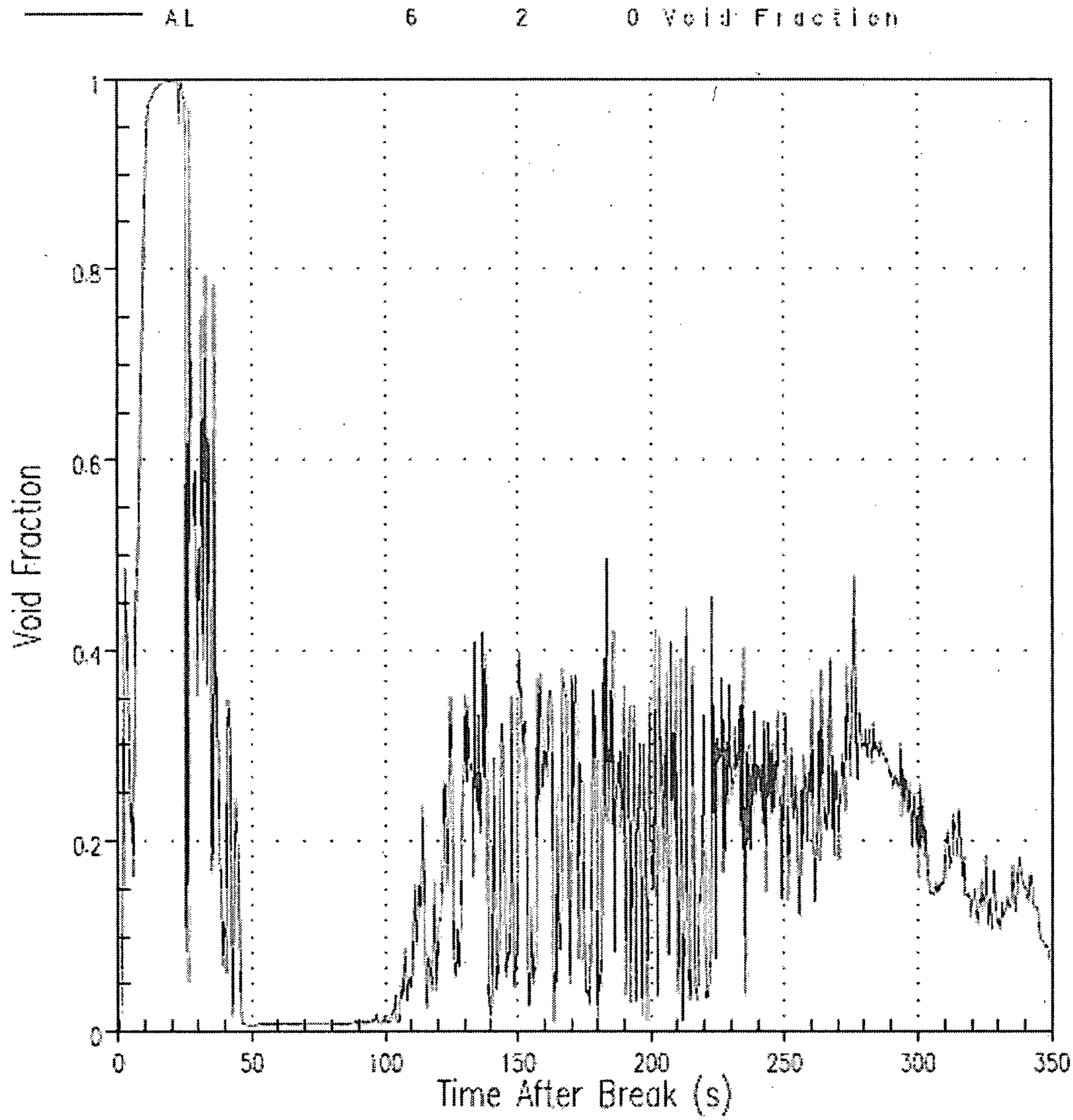
PTN Limiting PCT Case

Figure b9: Void Fraction from Adjacen to Broken Loop Downcomer Channel 56
Lower Plenum – Section 2 – at Level 2



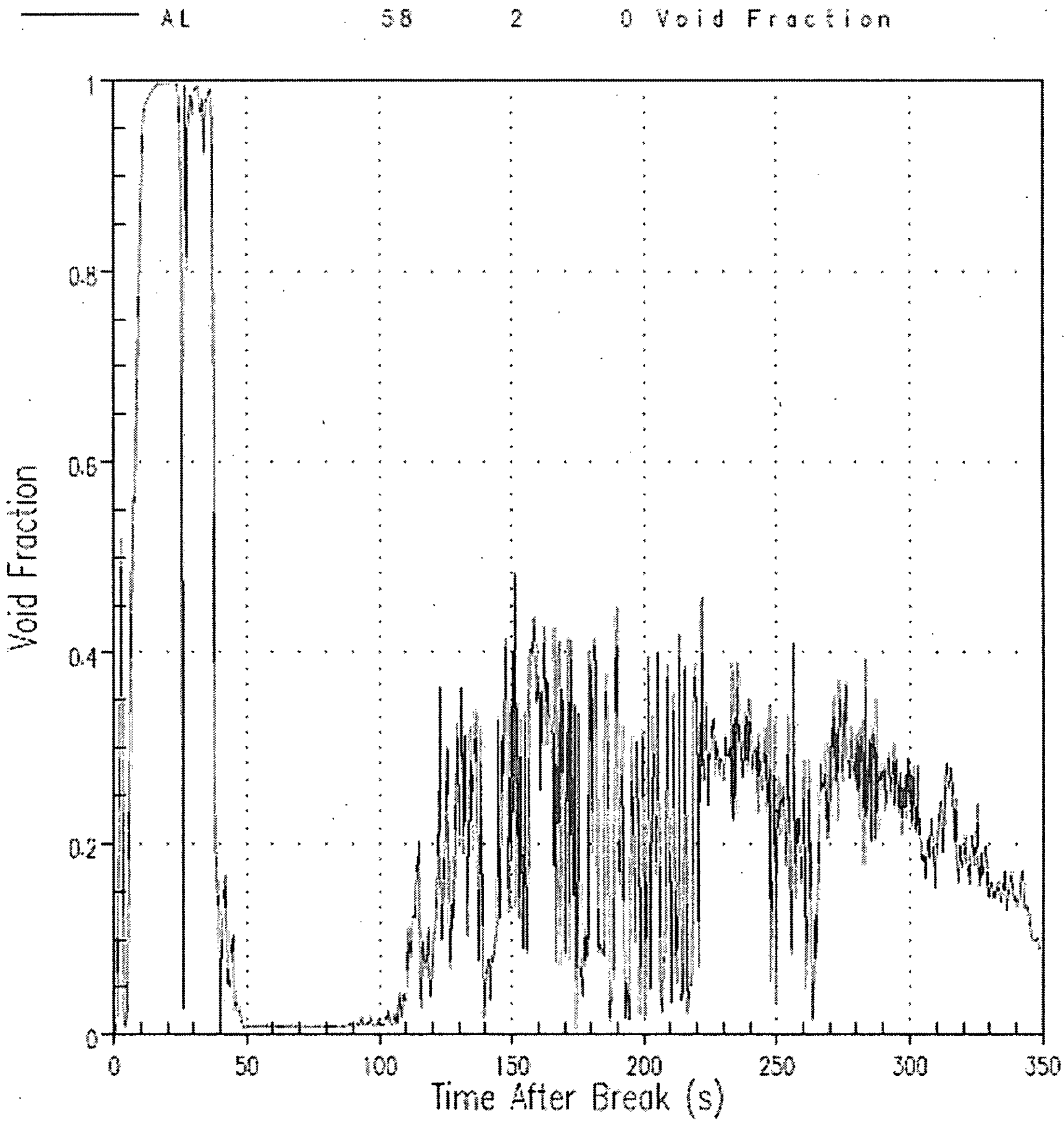
PTN Limiting PCT Case

Figure b10: Void Fraction from Intact Loop Downcomer Channel 6 Core – Section 3 – at Bottom of Section



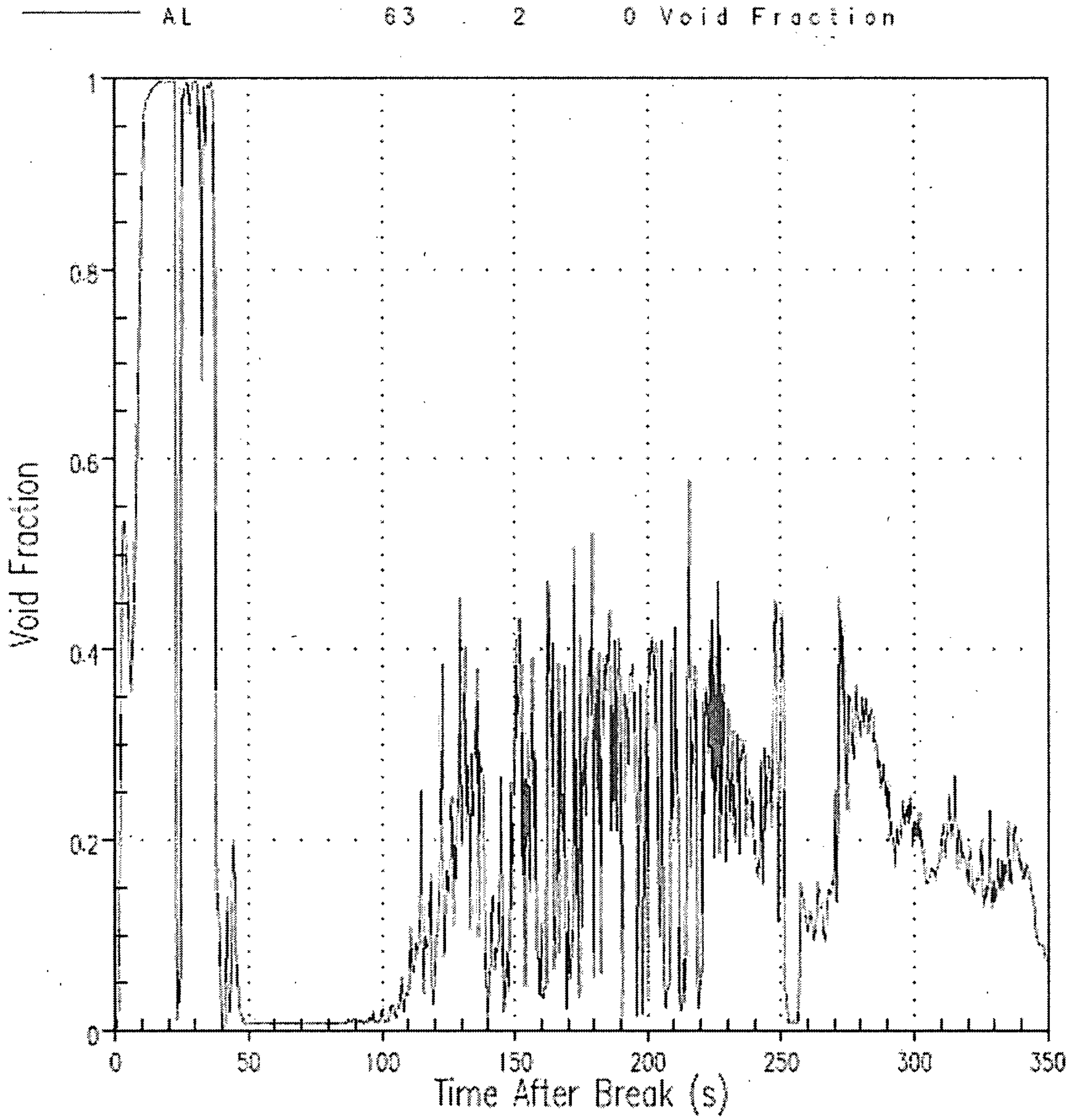
PTN Limiting PCT Case

Figure b11: Void Fraction from Intact Loop Downcomer Channel 58 Core – Section 3 – at Bottom of Section



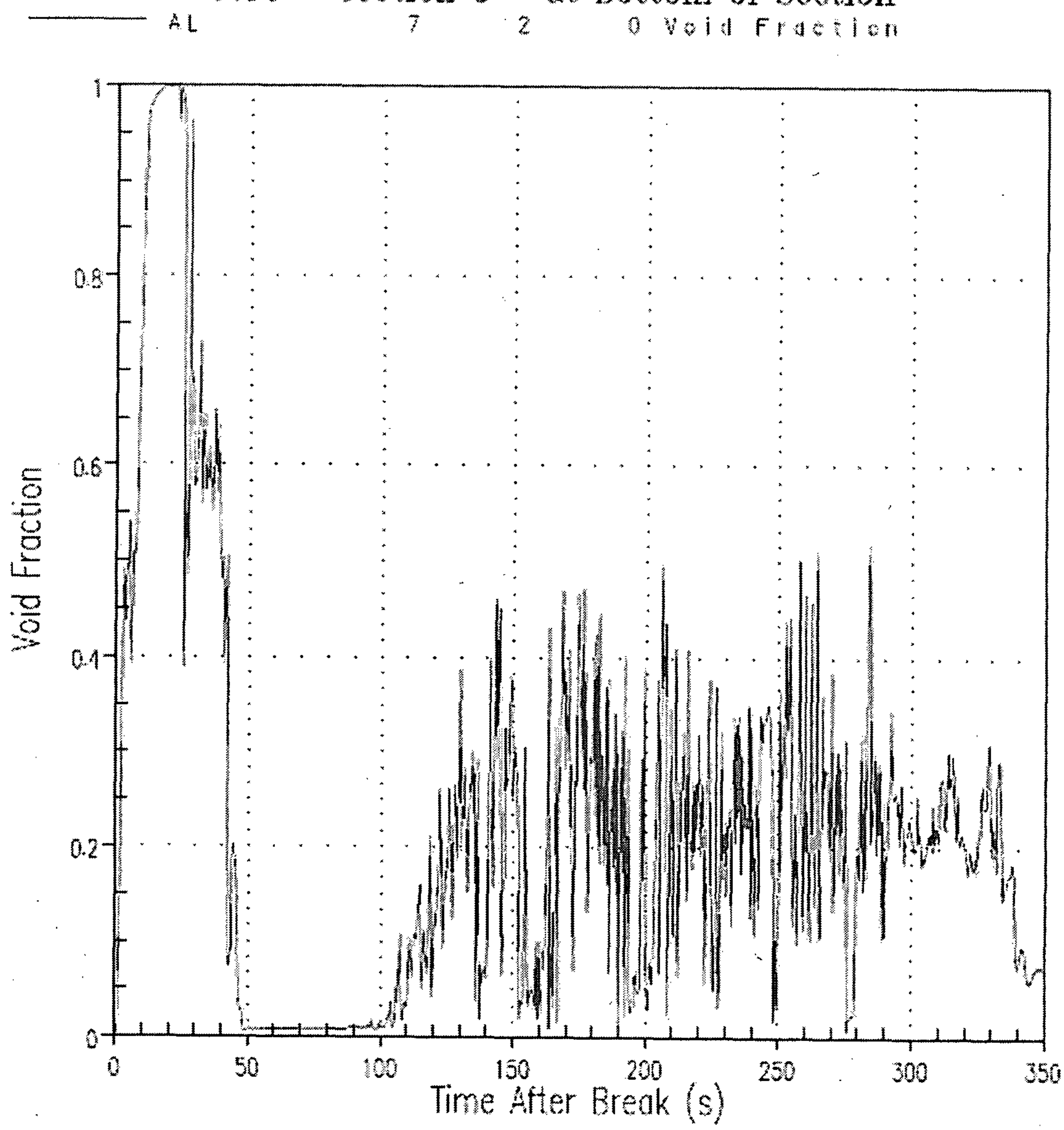
PTN Limiting PCT Case

Figure b12: Void Fraction from Intact Loop Downcomer Channel 63 Core – Section 3 – at Bottom of Section



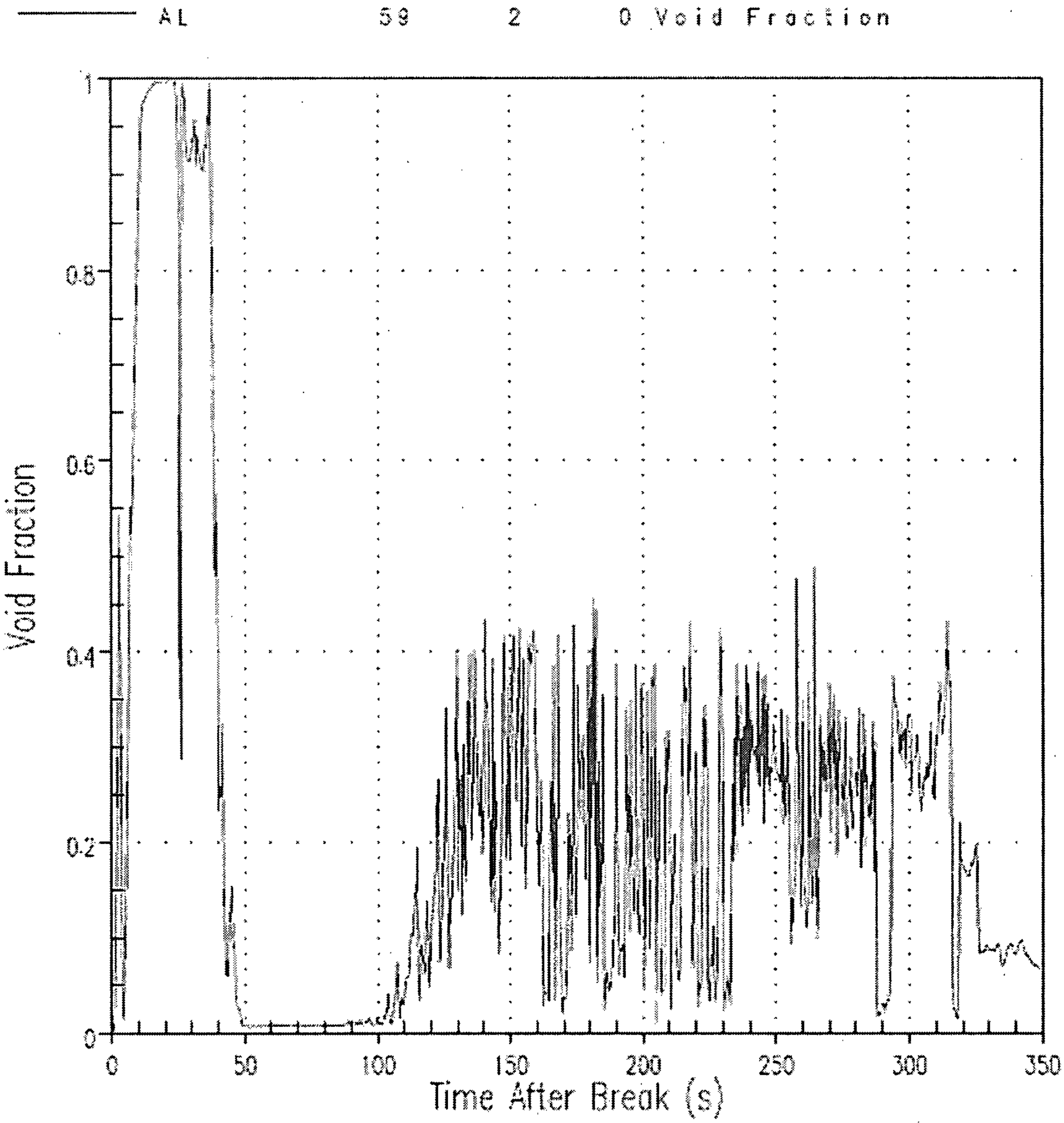
PTN Limiting PCT Case

Figure b13: Void Fraction from Intact Loop Downcomer Channel 7
Core - Section 3 - at Bottom of Section



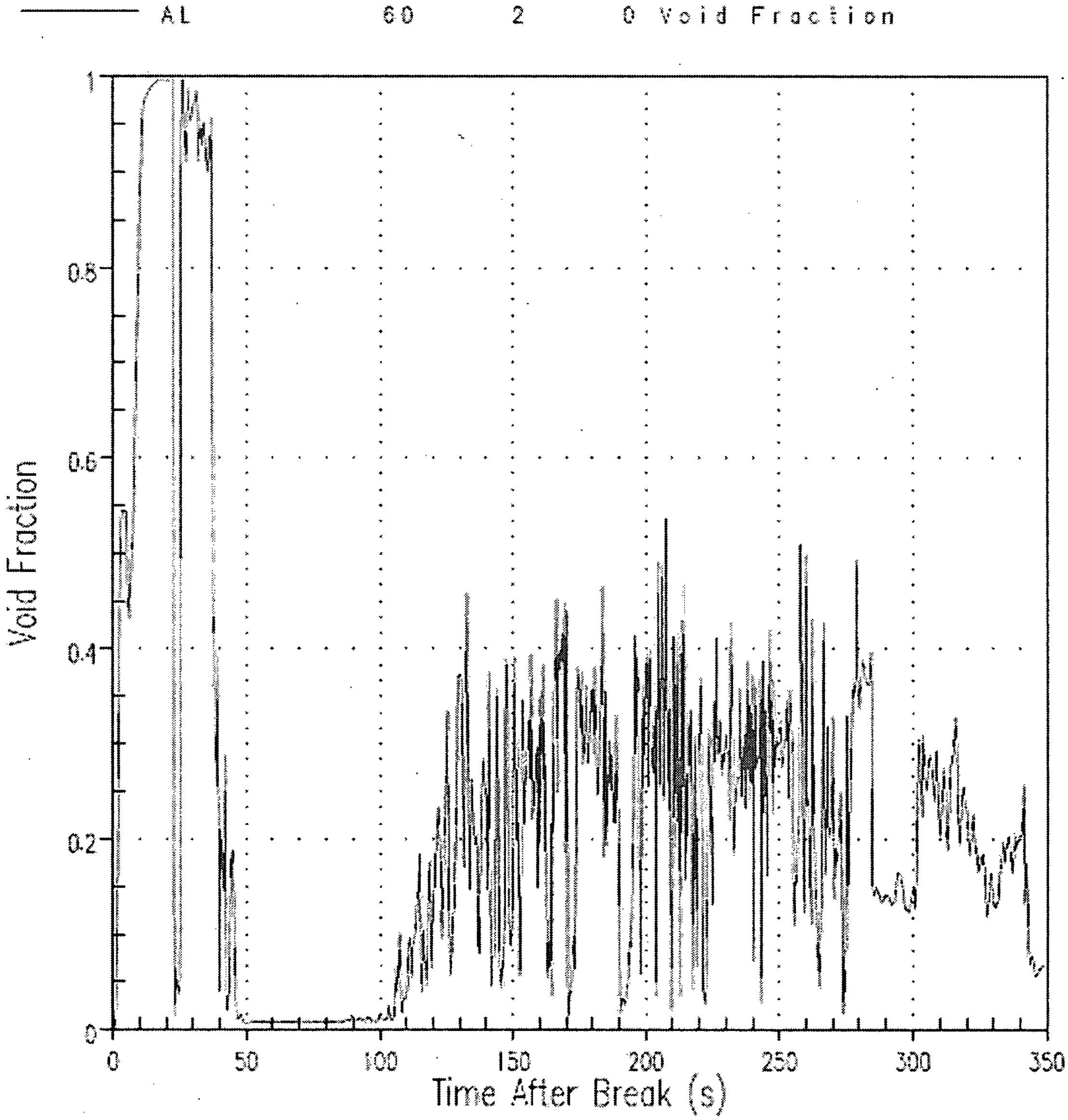
PTN Limiting PCT Case

Figure b14: Void Fraction from Intact Loop Downcomer Channel 59
Core – Section 3 – at Bottom of Section



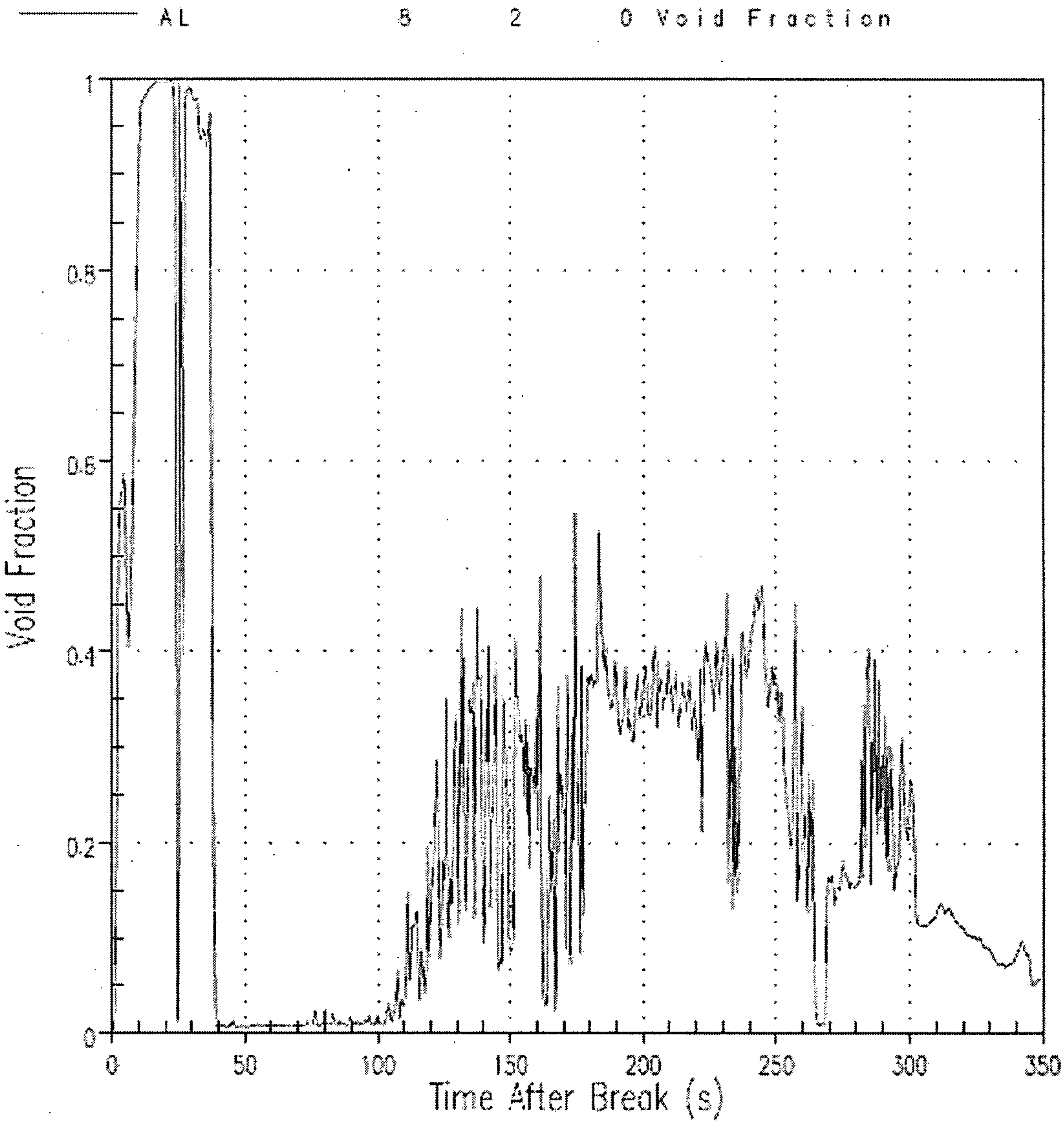
PTN Limiting PCT Case

Figure b15: Void Fraction from Intact Loop Downcomer Channel 60
Core – Section 3 – at Bottom of Section



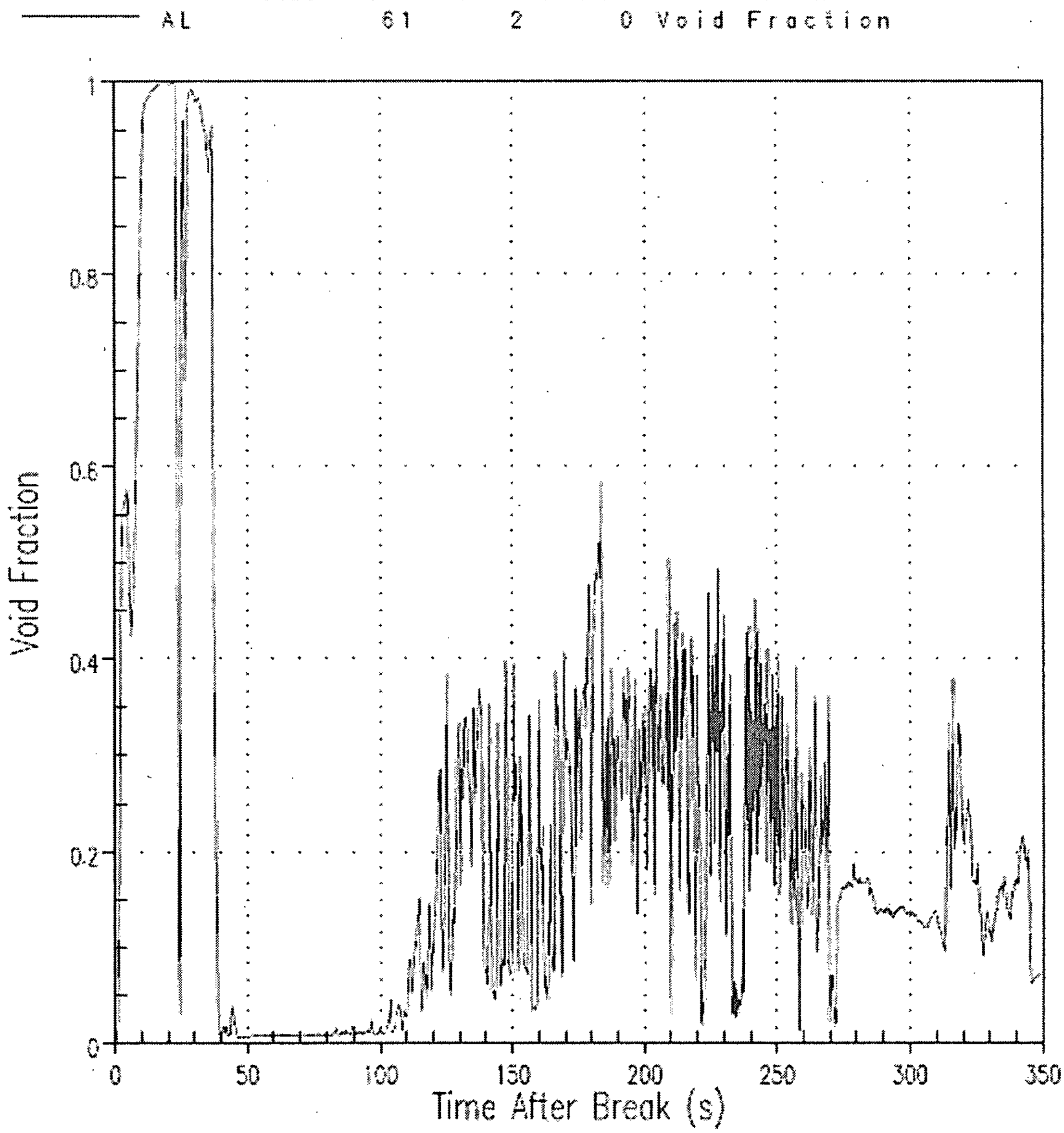
PTN Limiting PCT Case

Figure b16: Void Fraction from Broken Loop Downcomer Channel 8 Core – Section 3 – at Bottom of Section



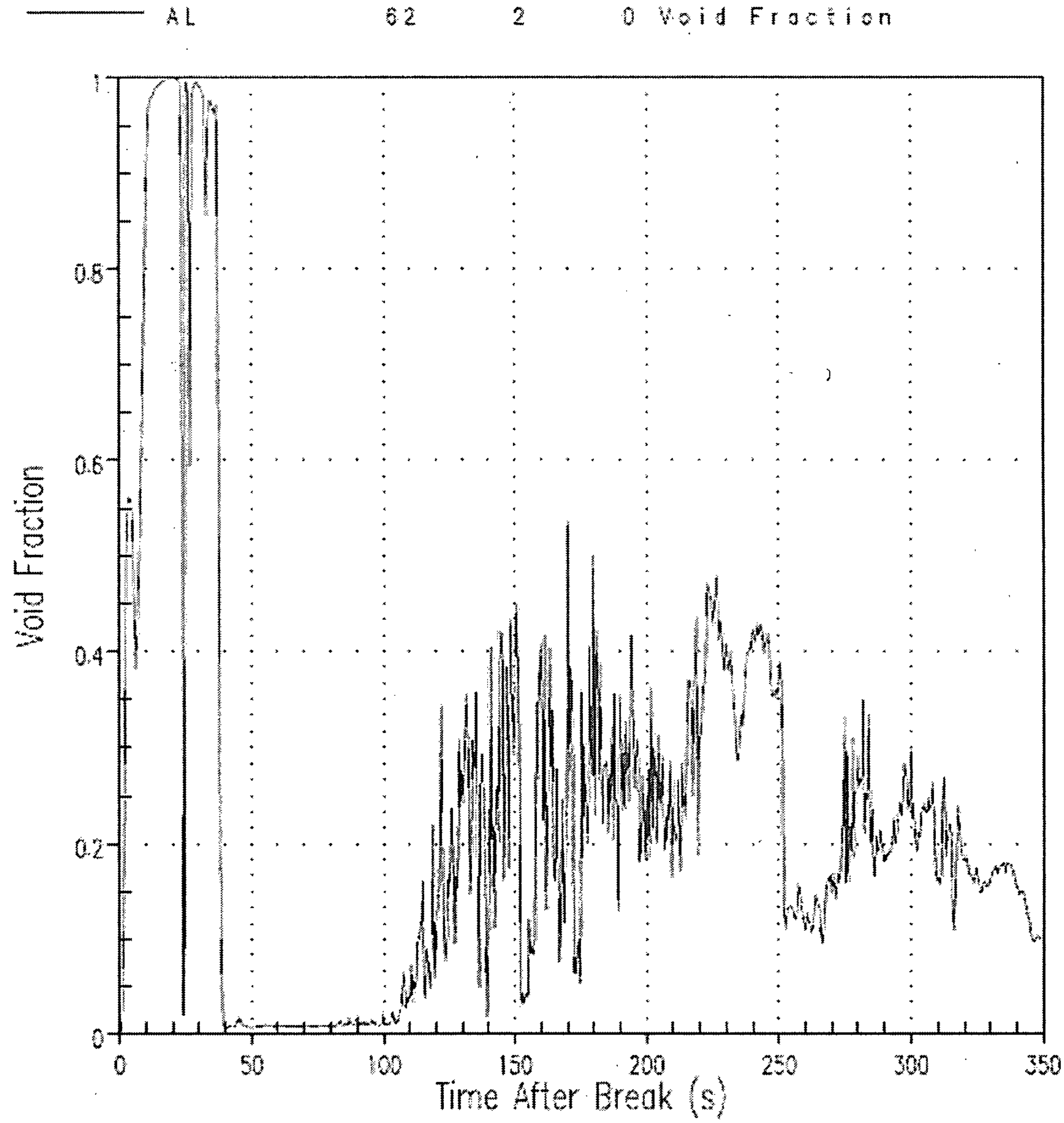
PTN Limiting PCT Case

Figure b17: Void Fraction from Adjacen to Broken Loop Downcomer Channel 61
Core – Section 3 – at Bottom of Section



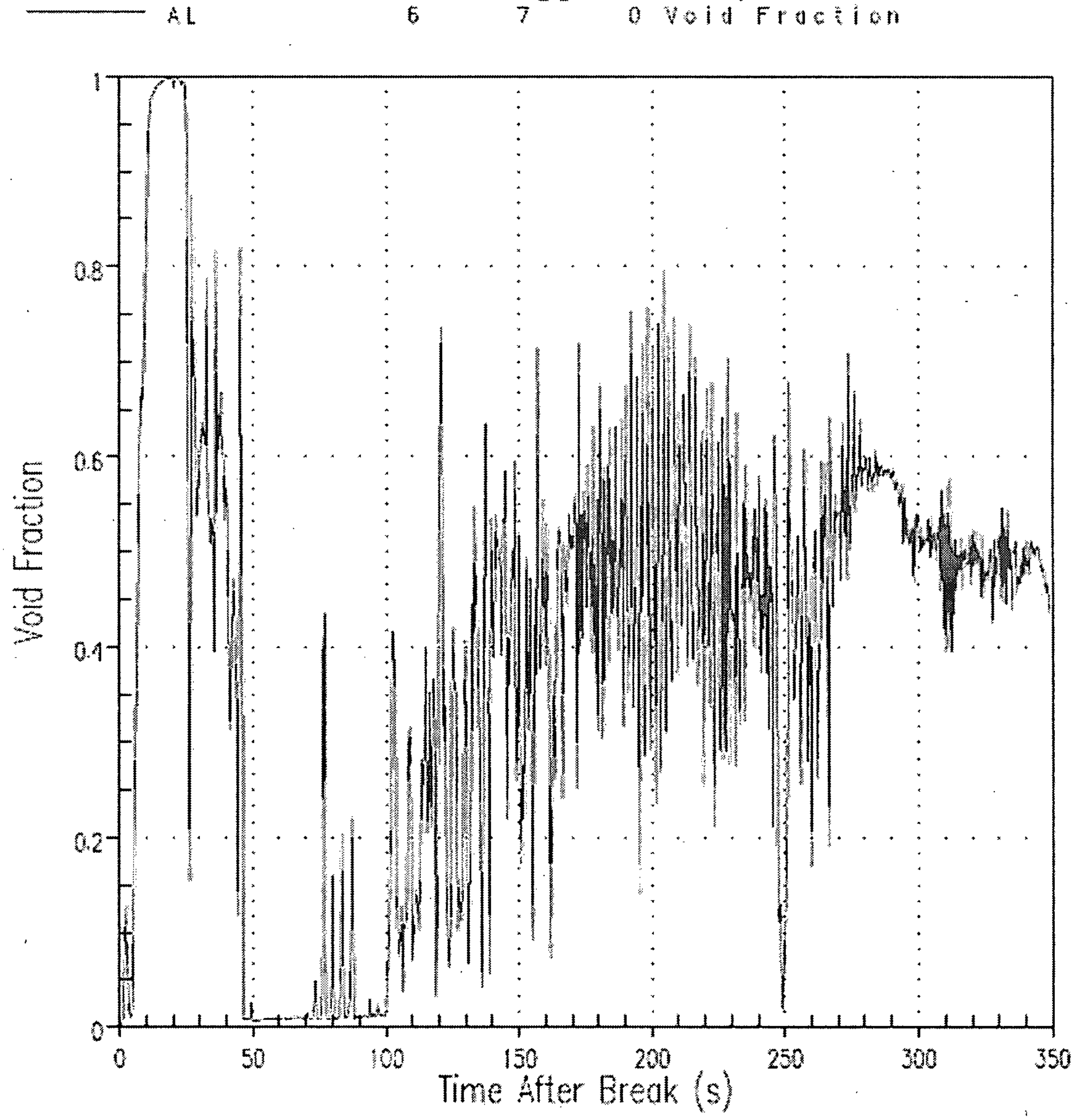
PTN Limiting PCT Case

**Figure b18: Void Fraction from Adjacen to Broken Loop Downcomer Channel 62
Core - Section 3 - at Bottom of Section**



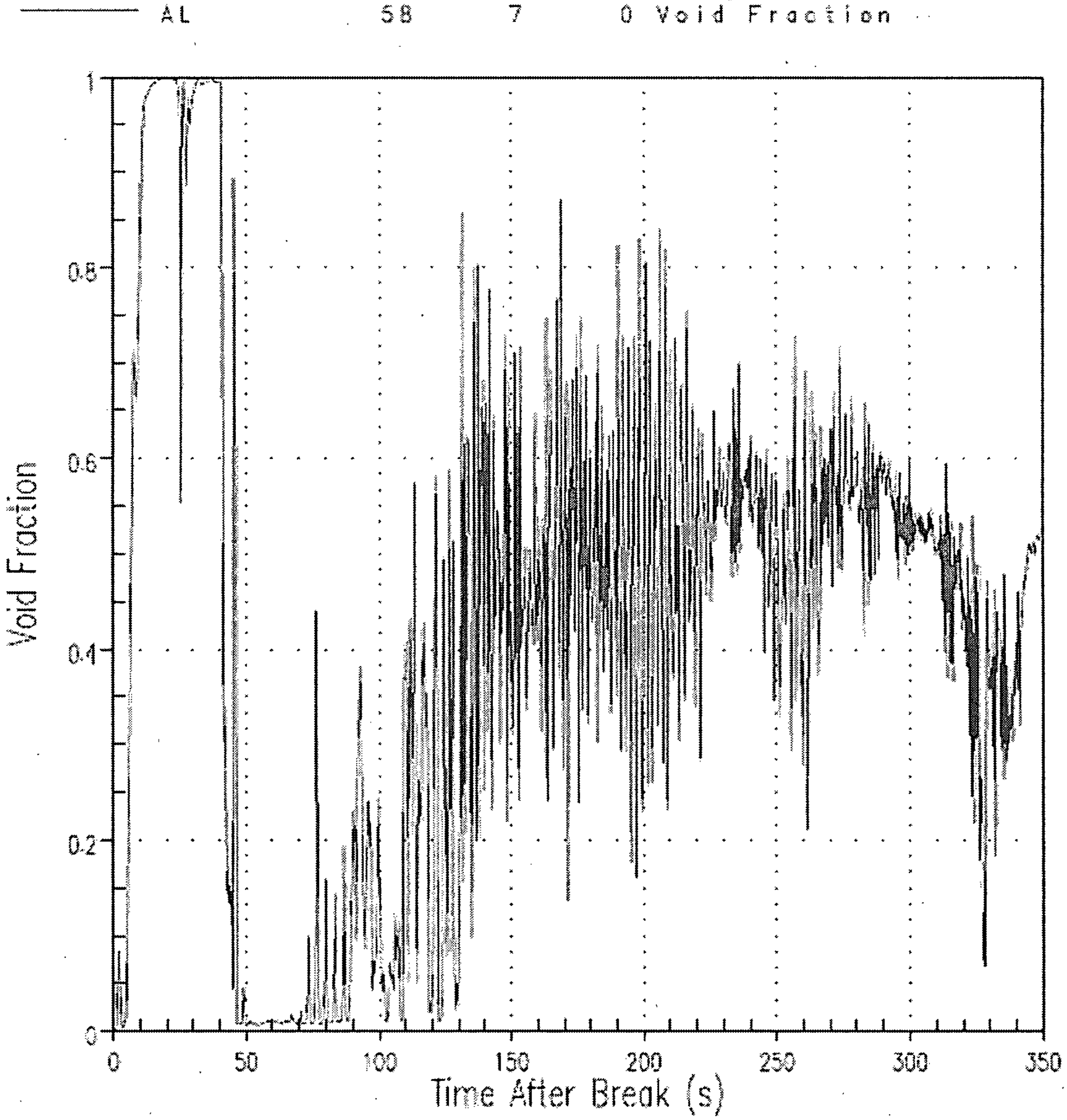
PTN Limiting PCT Case

Figure b19: Void Fraction from Intact Loop Downcomer Channel 6
Core - Section 3 - at Approximately Mid-Section



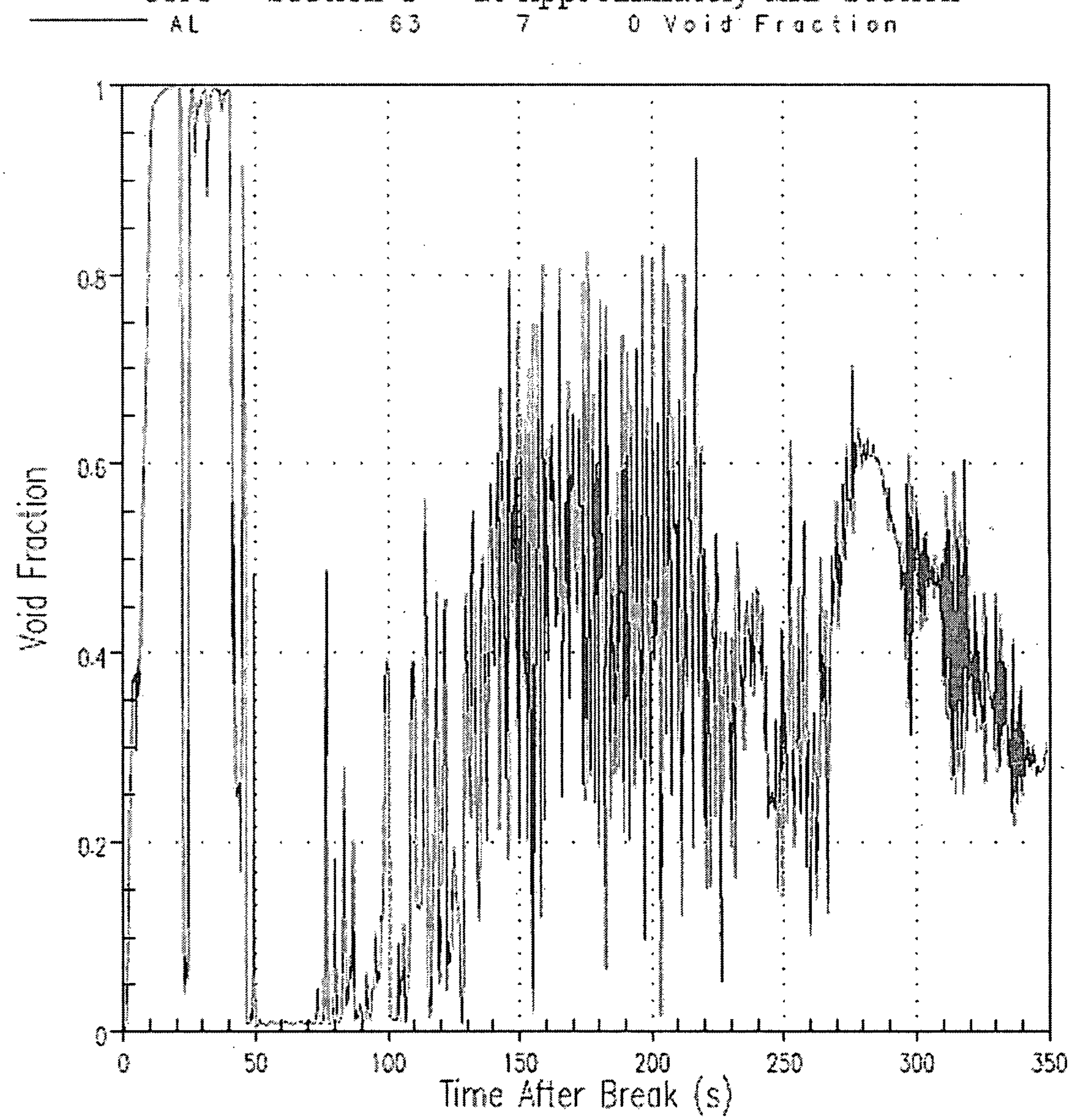
PTN Limiting PCT Case

Figure b20 Void Fraction from Intact Loop Downcomer Channel 58
Core – Section 3 – at Approximately Mid-Section



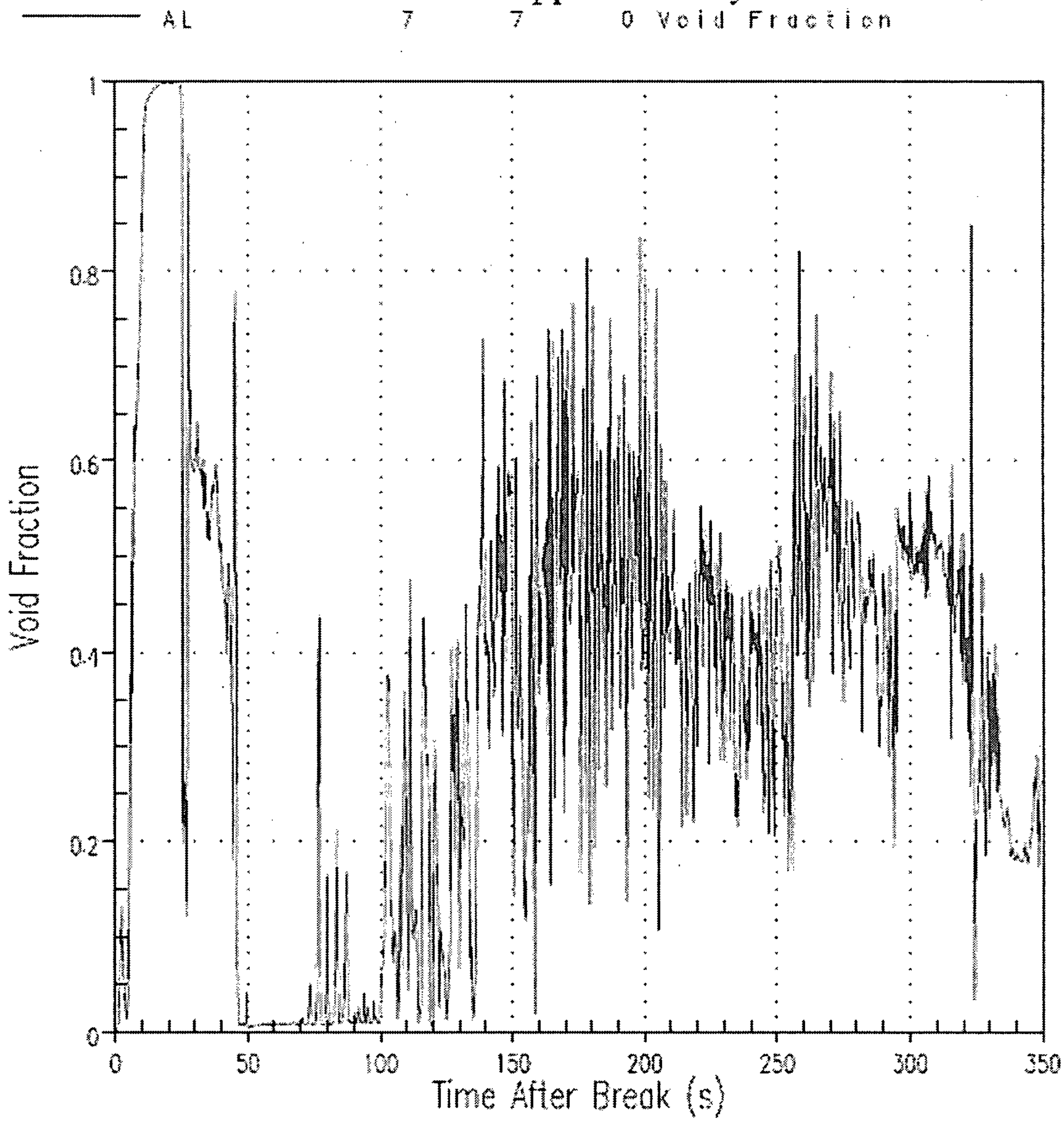
PTN Limiting PCT Case

Figure b21: Void Fraction from Intact Loop Downcomer Channel 63 Core - Section 3 - at Approximately Mid-Section



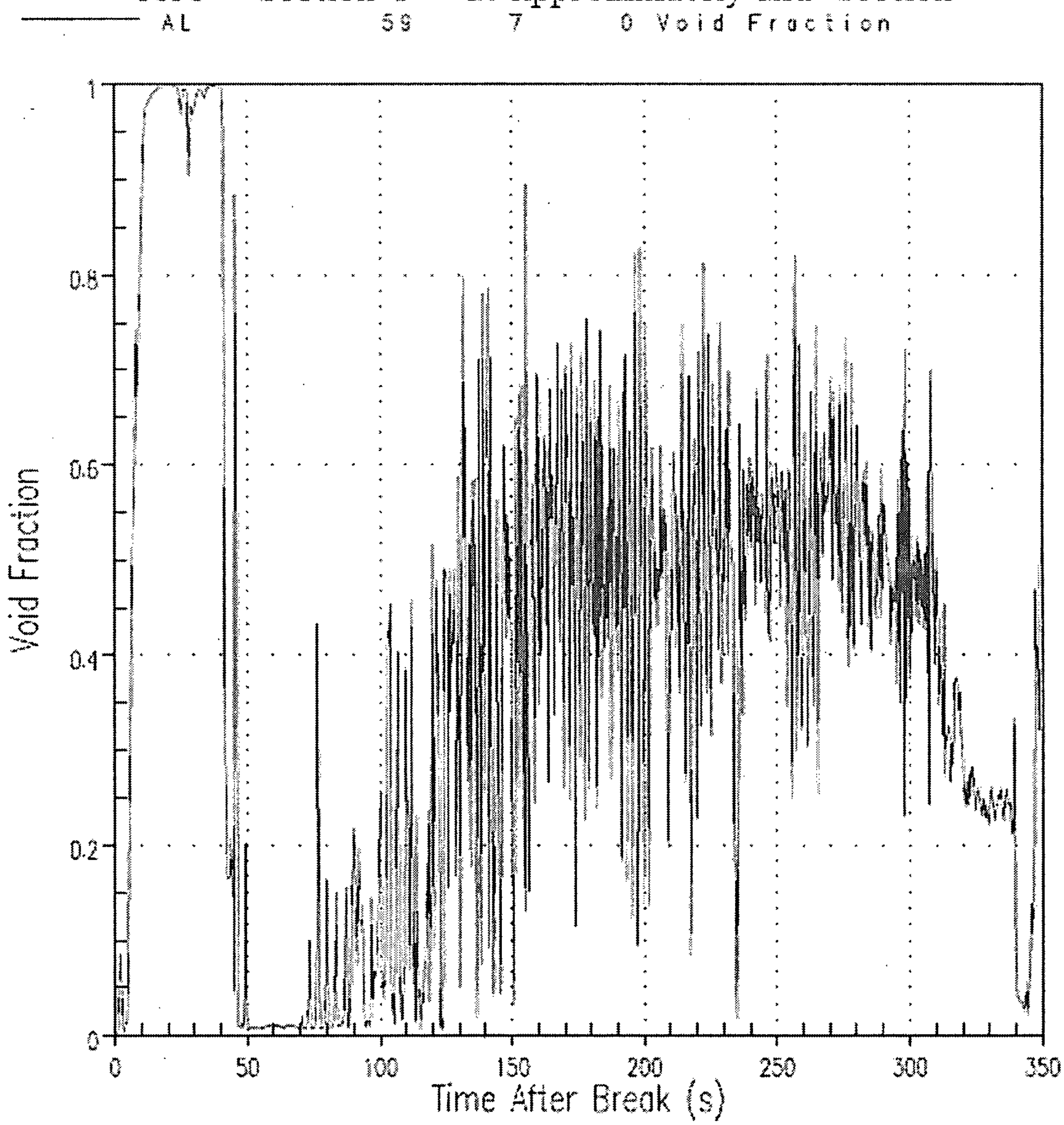
PTN Limiting PCT Case

Figure b22: Void Fraction from Intact Loop Downcomer Channel 7 Core - Section 3 - at Approximately Mid-Section



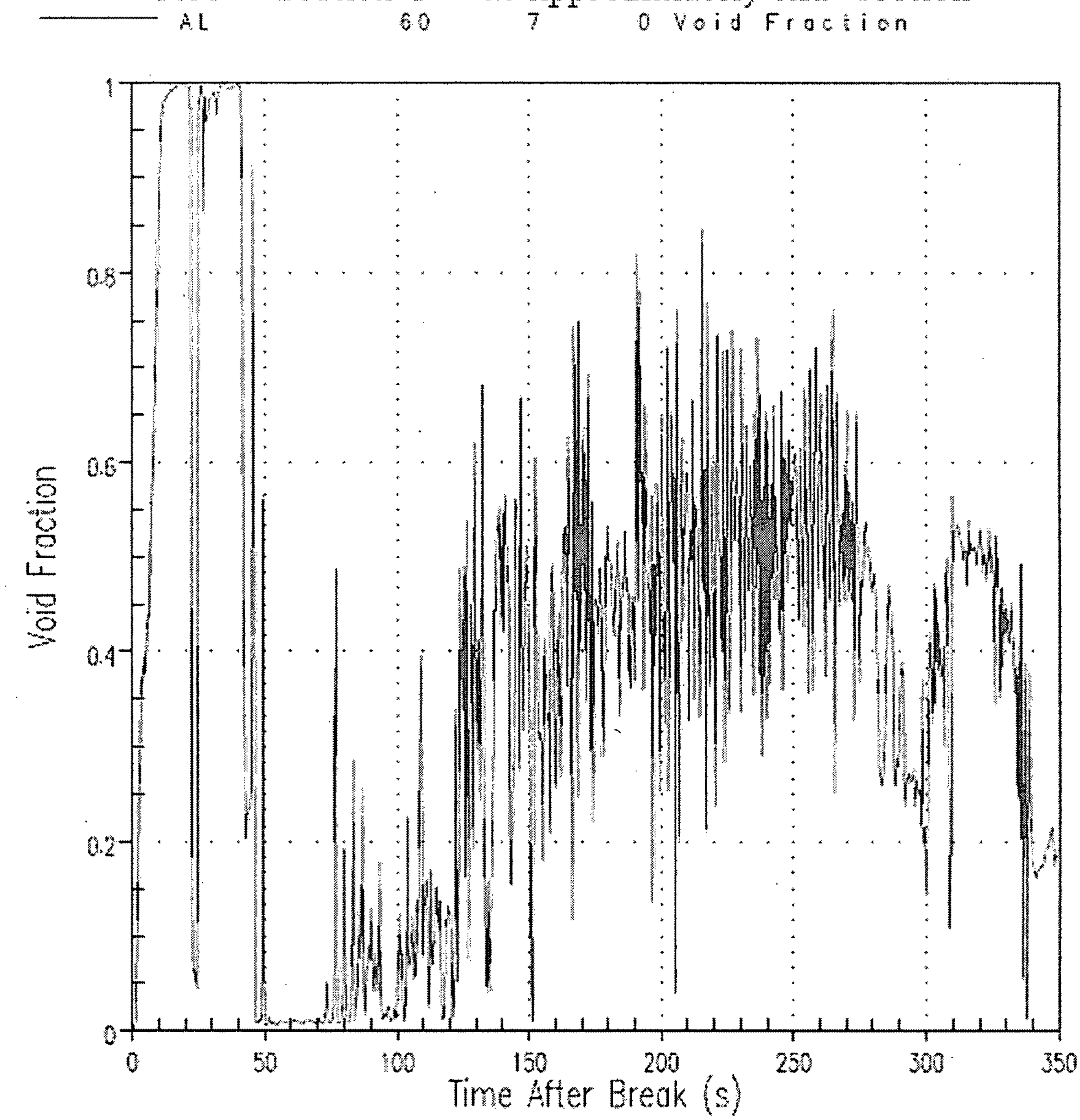
PTN Limiting PCT Case

Figure b23: Void Fraction from Intact Loop Downcomer Channel 59 Core – Section 3 – at Approximately Mid-Section



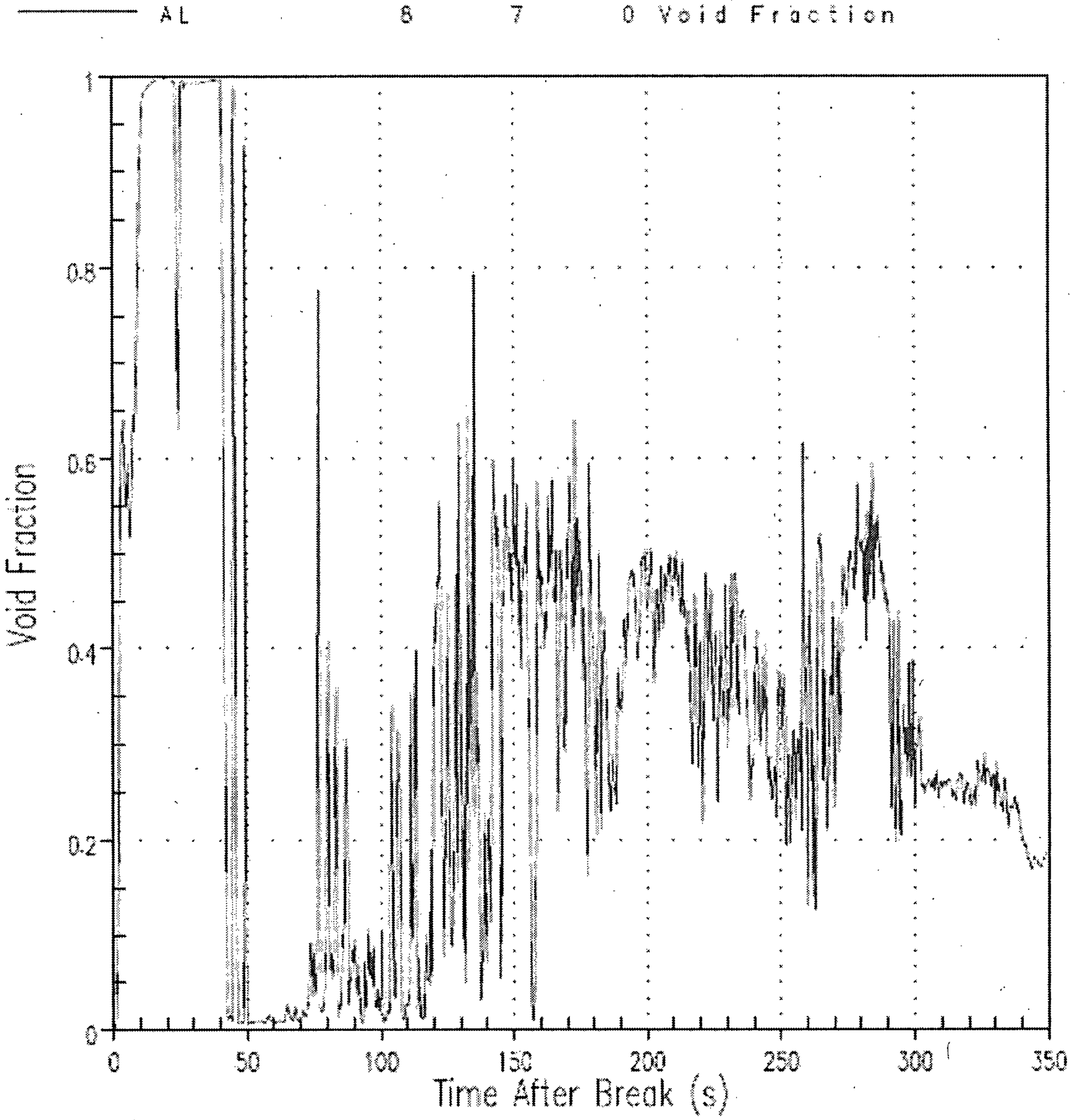
PTN Limiting PCT Case

Figure b24: Void Fraction from Intact Loop Downcomer Channel 60 Core - Section 3 - at Approximately Mid-Section



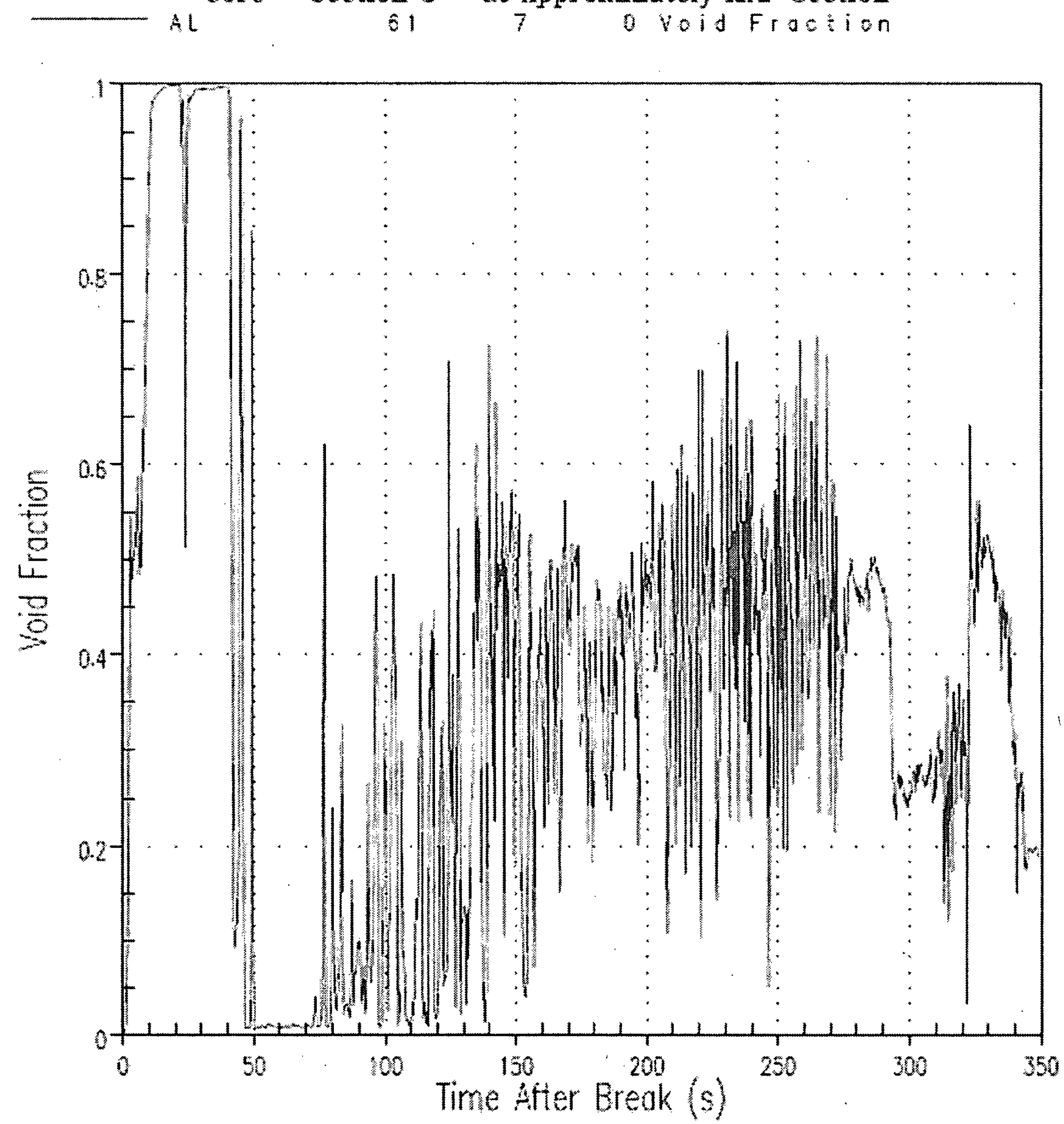
PTN Limiting PCT Case

Figure b25: Void Fraction from Broken Loop Downcomer Channel 8 Core – Section 3 – at Approximately Mid-Section



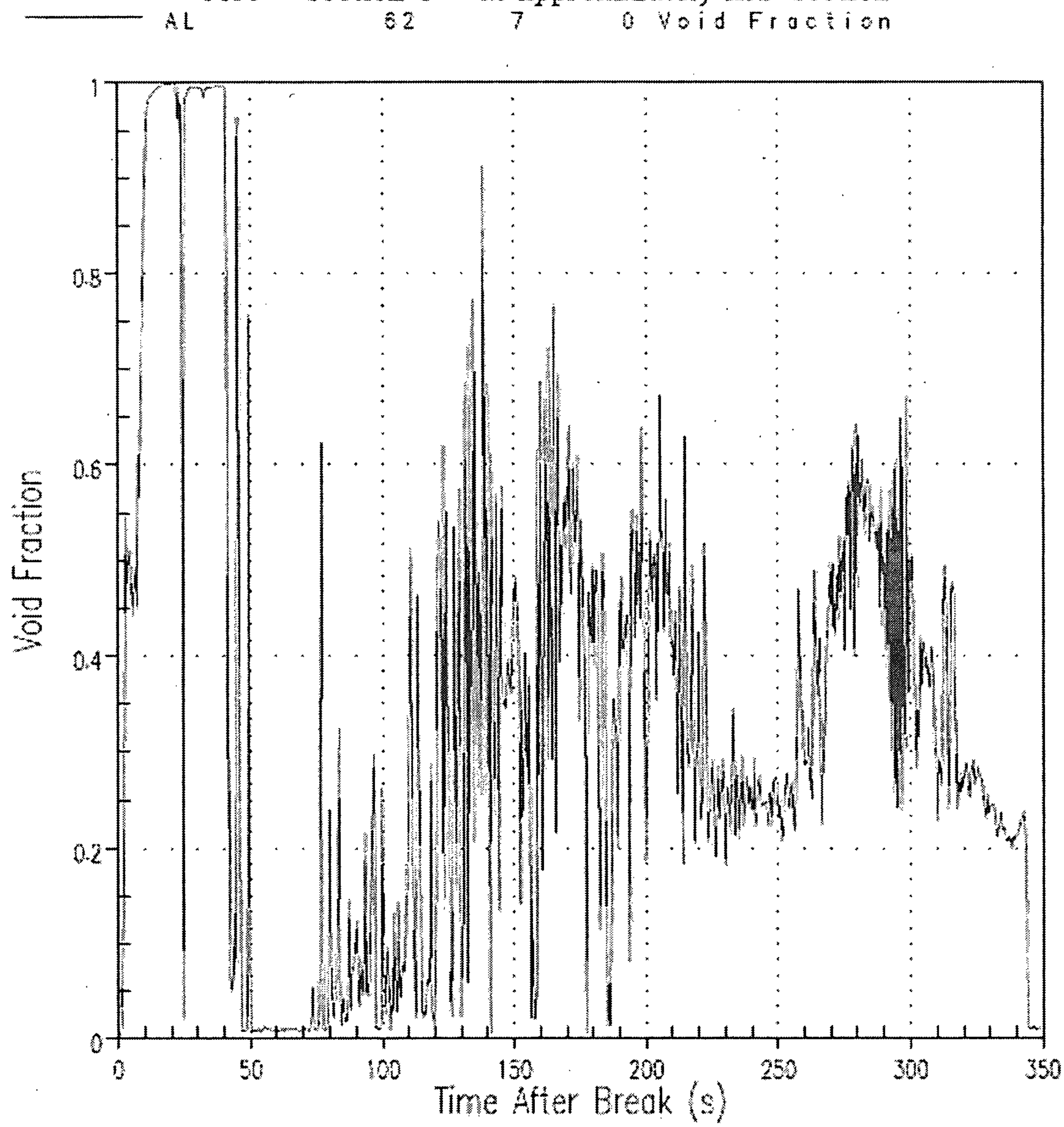
PTN Limiting PCT Case

Figure b26: Void Fraction from Adjacen to Broken Loop Downcomer Channel 61 Core - Section 3 - at Approximately Mid-Section



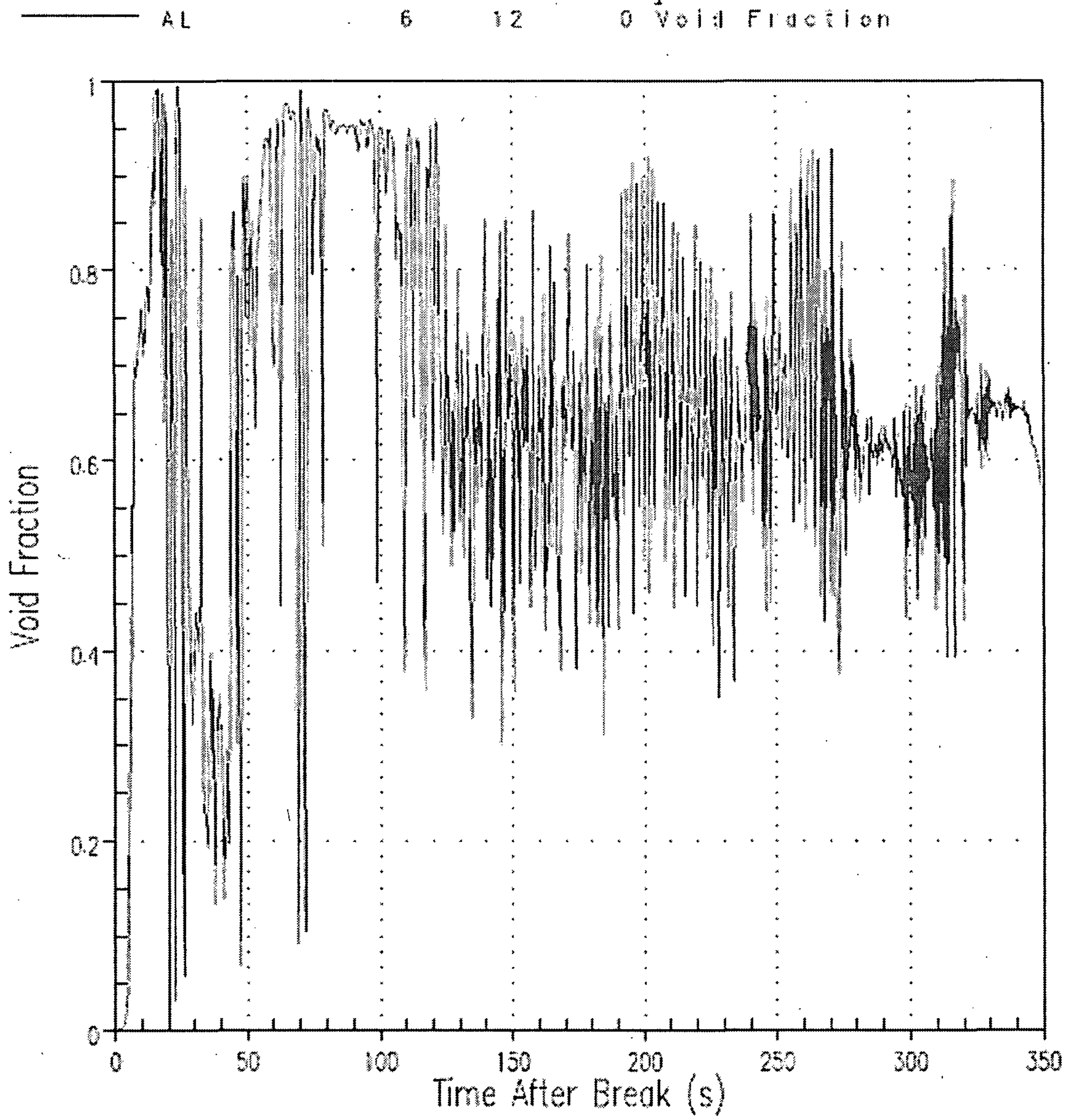
PTN Limiting PCT Case

Figure b27: Void Fraction from Adjacen to Broken Loop Downcomer Channel 62
Core - Section 3 - at Approximately Mid-Section



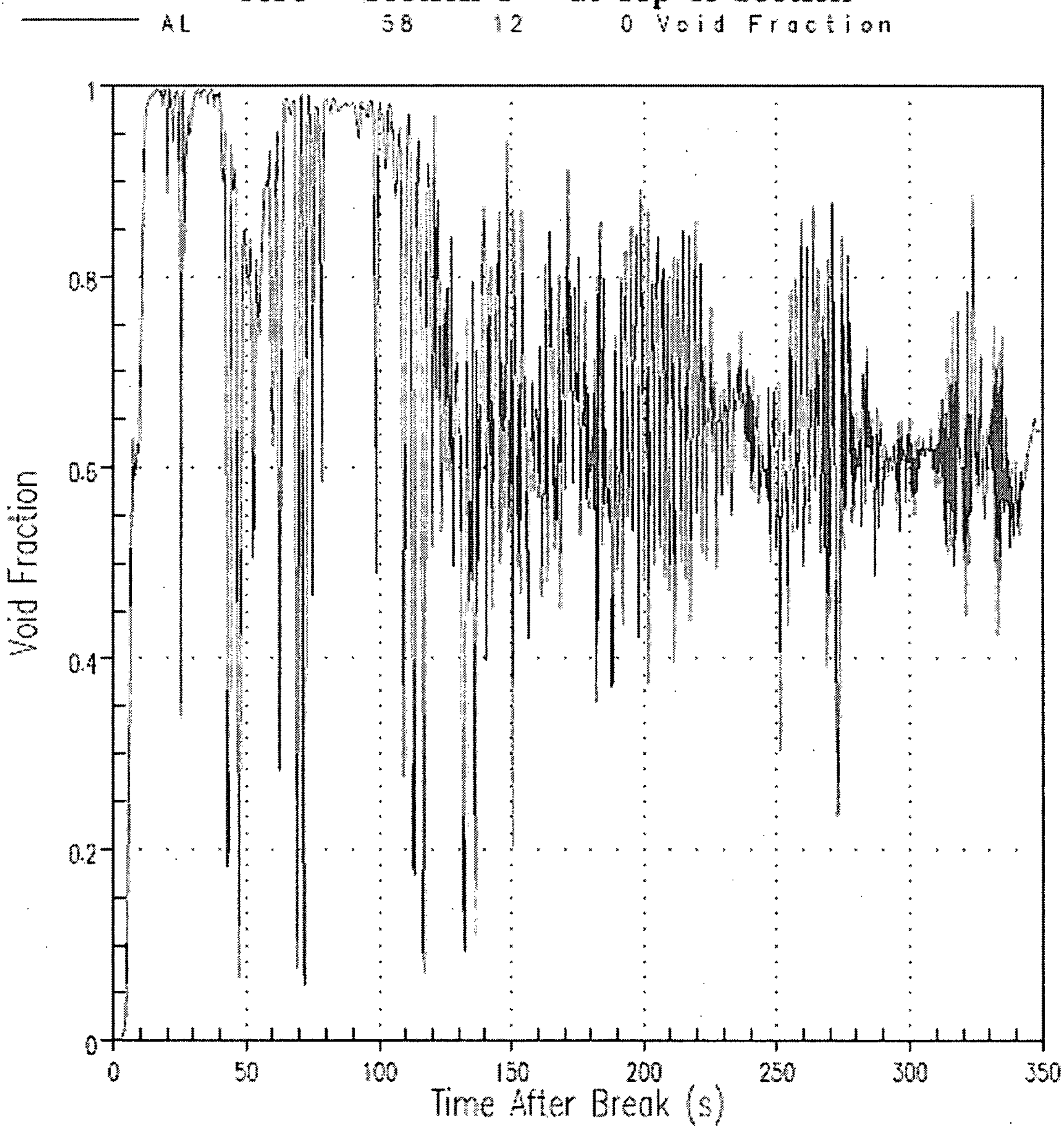
PTN Limiting PCT Case

Figure b28: Void Fraction from Intact Loop Downcomer Channel 6
Core - Section 3 - at Top of Section



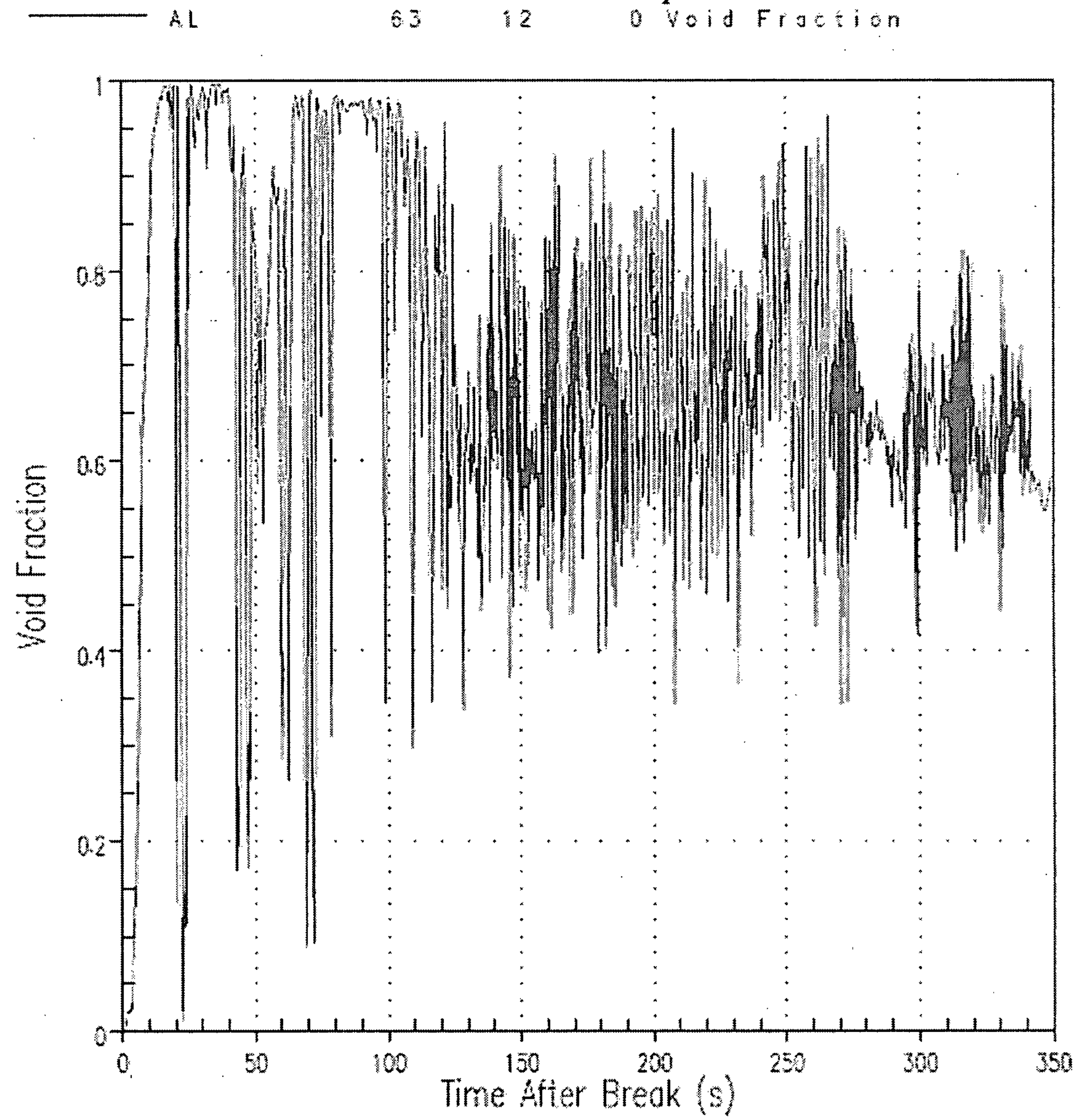
PTN Limiting PCT Case

Figure b29: Void Fraction from Intact Loop Downcomer Channel 58 Core - Section 3 - at Top of Section



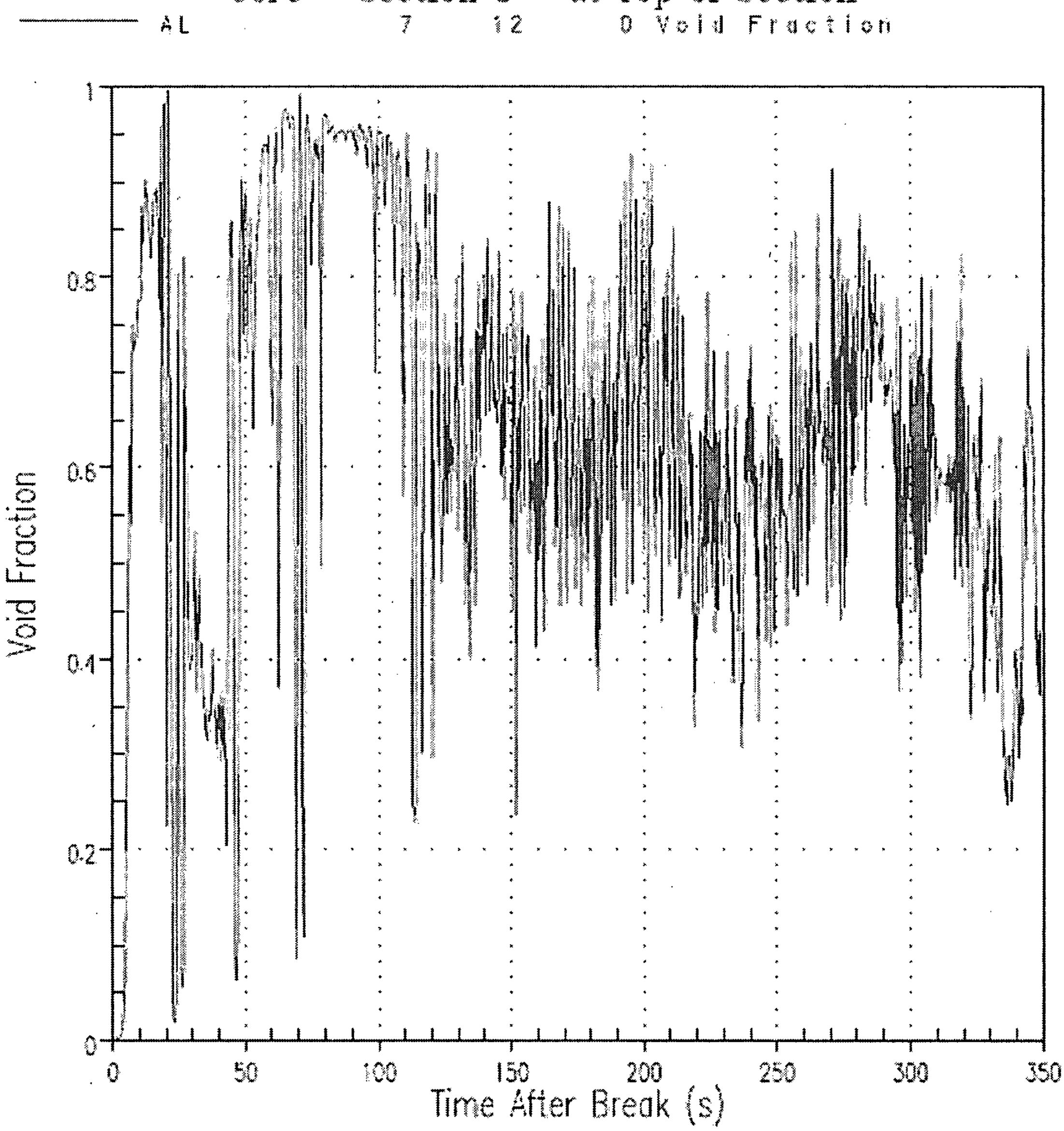
PTN Limiting PCT Case

Figure b30: Void Fraction from Intact Loop Downcomer Channel 63
Core – Section 3 – at Top of Section



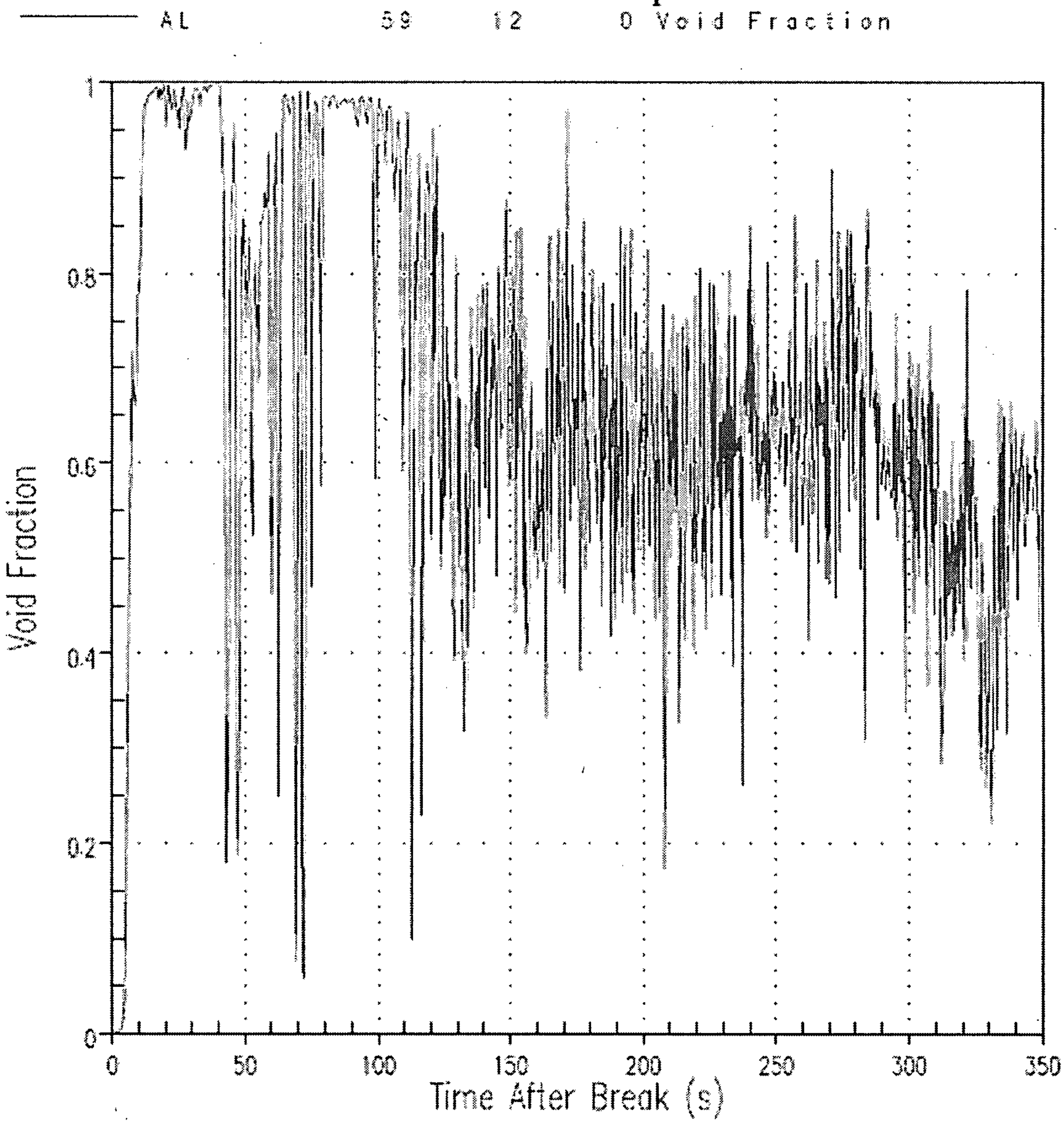
PTN Limiting PCT Case

Figure b31: Void Fraction from Intact Loop Downcomer Channel 7
Core – Section 3 – at Top of Section



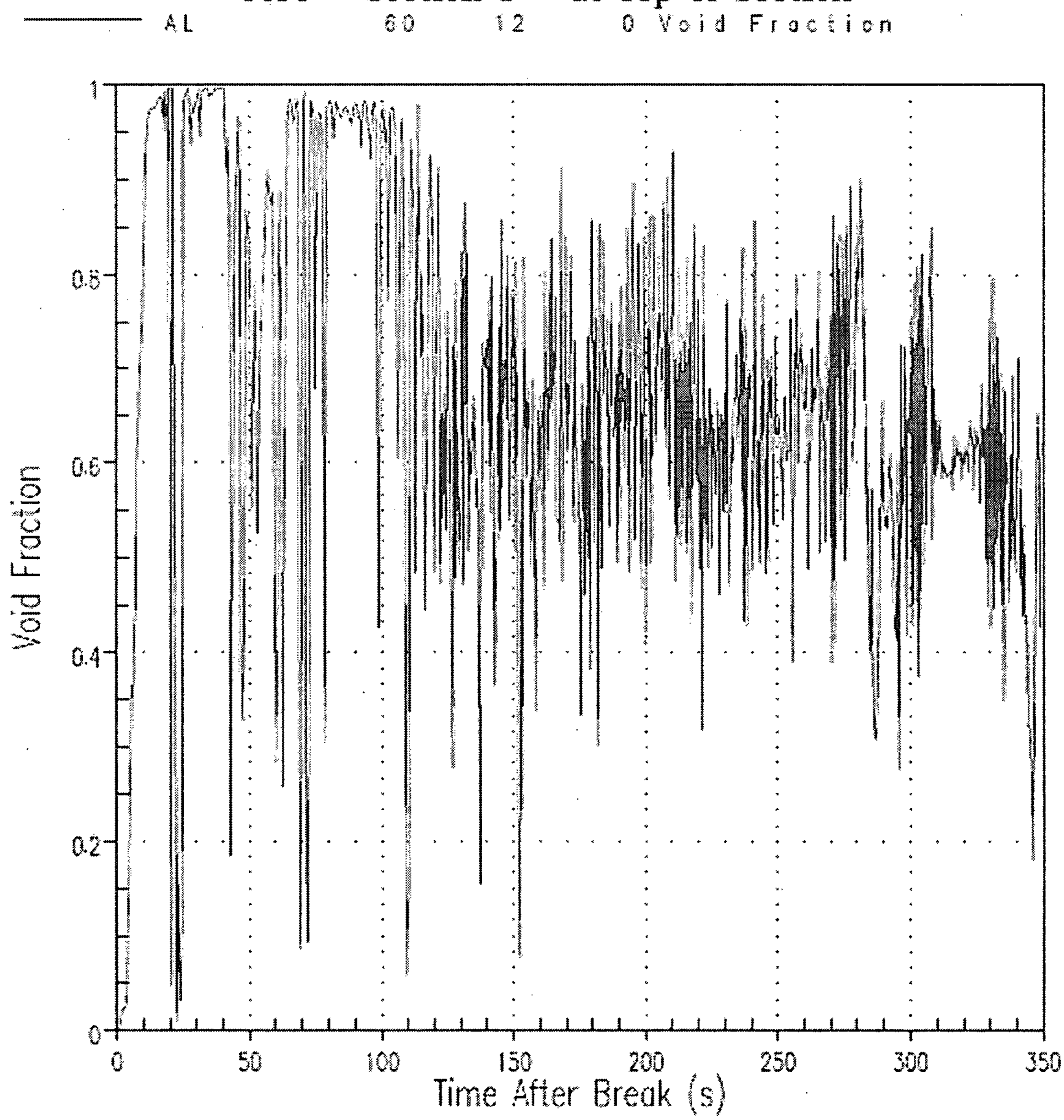
PTN Limiting PCT Case

Figure b32: Void Fraction from Intact Loop Downcomer Channel 59
Core - Section 3 - at Top of Section



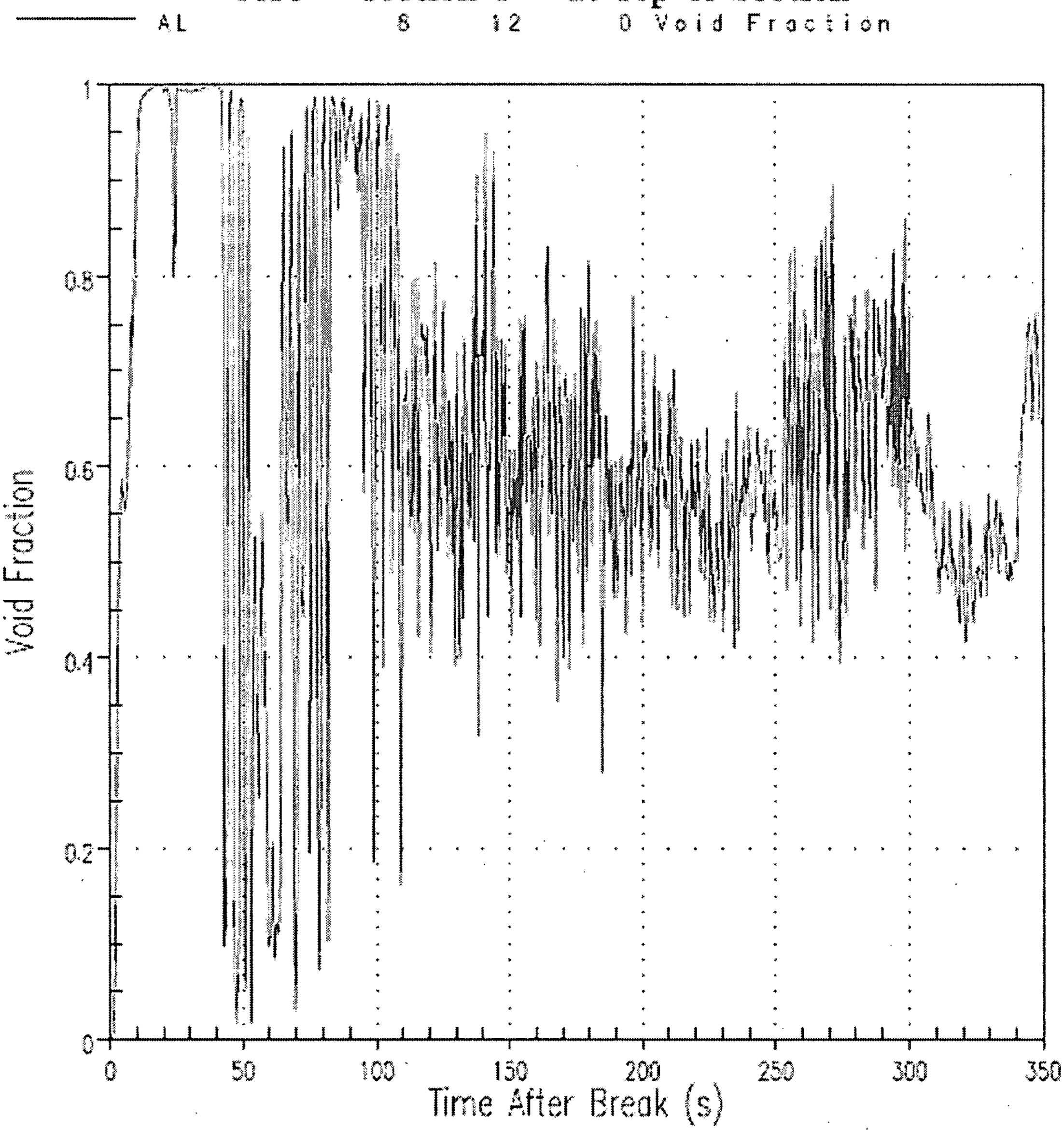
PTN Limiting PCT Case

Figure b33: Void Fraction from Intact Loop Downcomer Channel 60
Core – Section 3 – at Top of Section



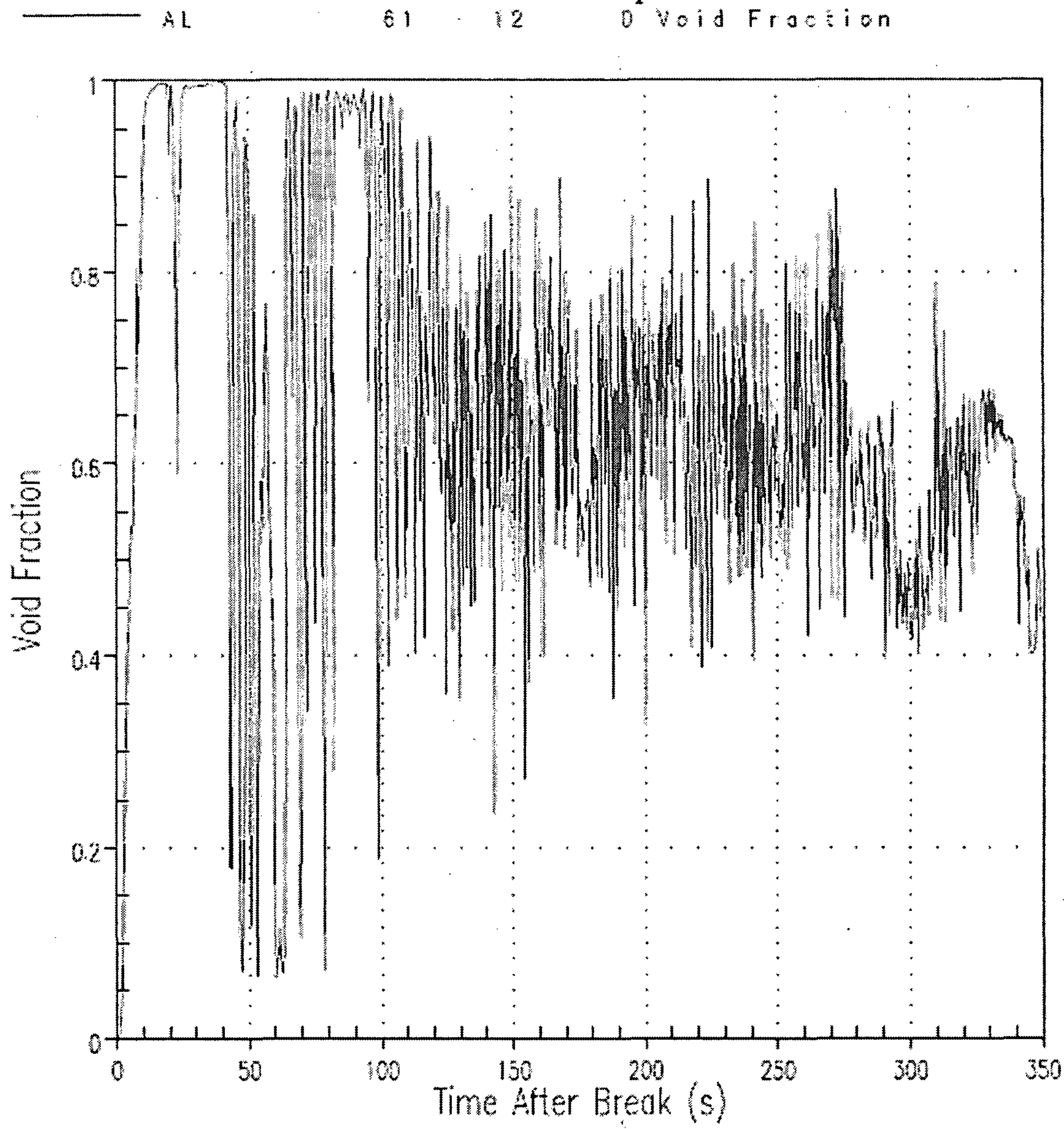
PTN Limiting PCT Case

Figure b34: Void Fraction from Broken Loop Downcomer Channel 8
Core – Section 3 – at Top of Section



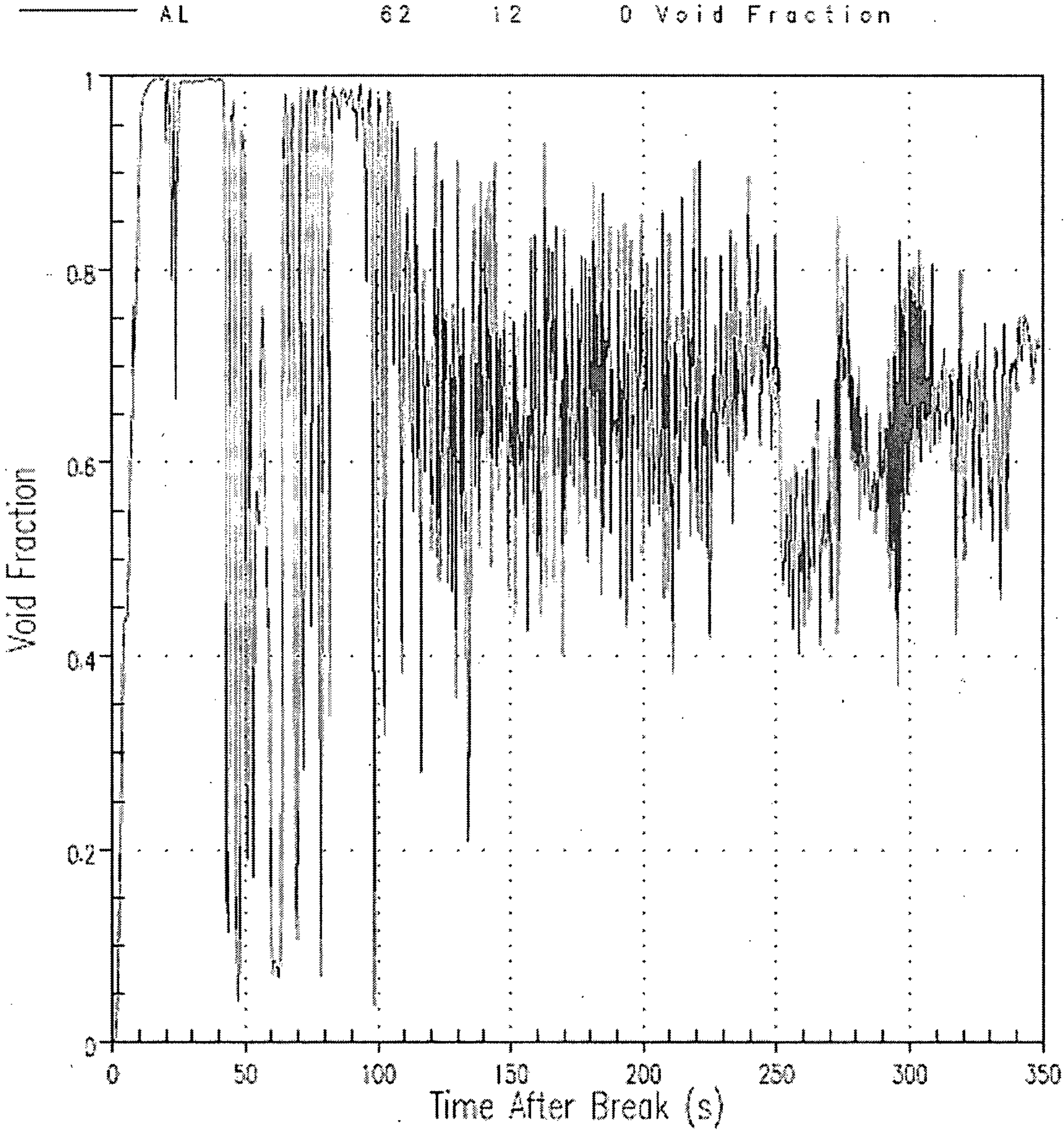
PTN Limiting PCT Case

Figure b35: Void Fraction from Adjacen to Broken Loop Downcomer Channel 61
Core - Section 3 - at Top of Section



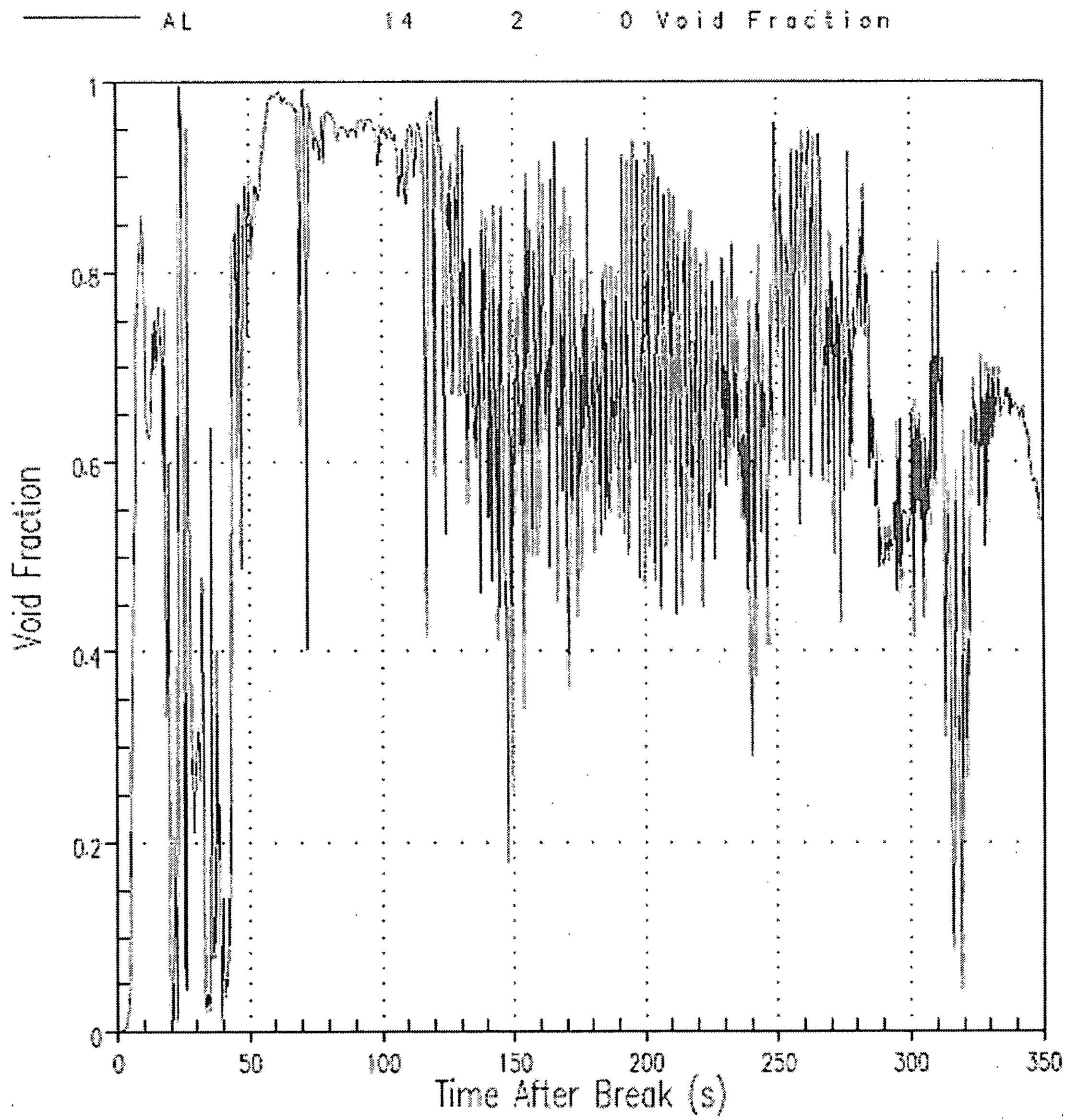
PTN Limiting PCT Case

Figure b36: Void Fraction from Adjacen to Broken Loop Downcomer Channel 62 Core - Section 3 - at Top of Section



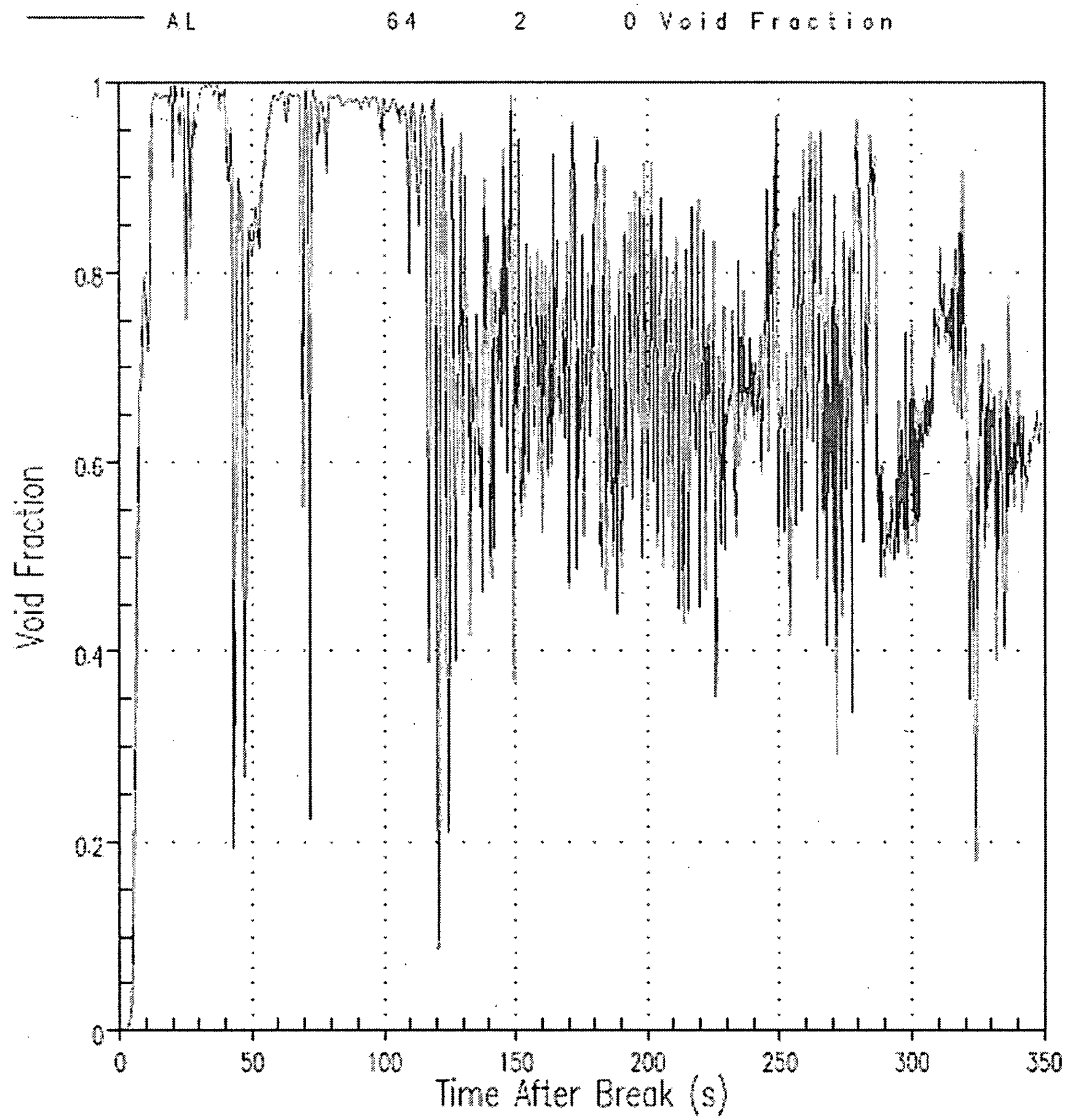
PTN Limiting PCT Case

Figure b37: Void Fraction from Intact Loop Downcomer Channel 14
CCFL – Section 4



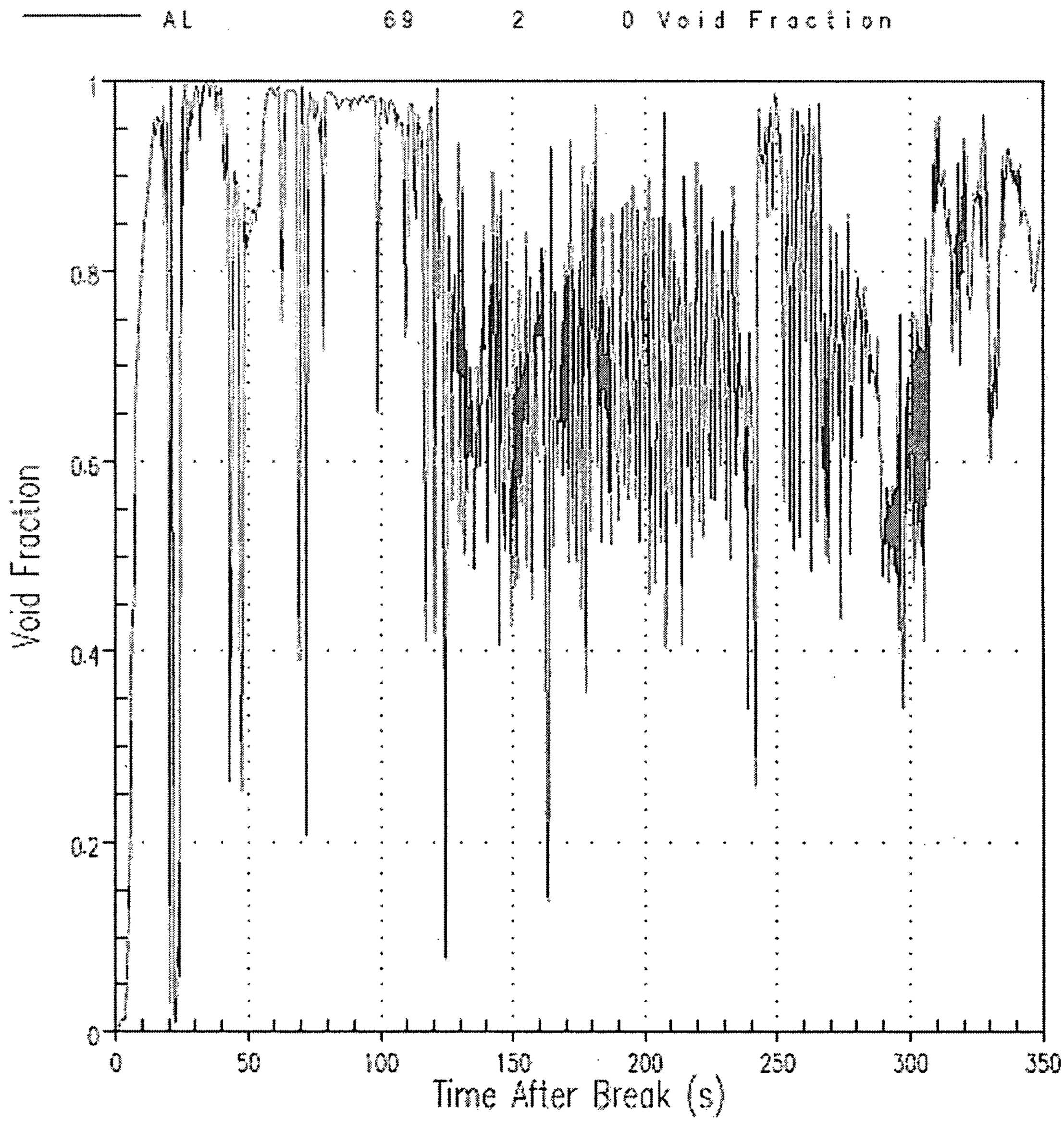
PTN Limiting PCT Case

Figure b38: Void Fraction from Intact Loop Downcomer Channel 64
CCFL – Section 4



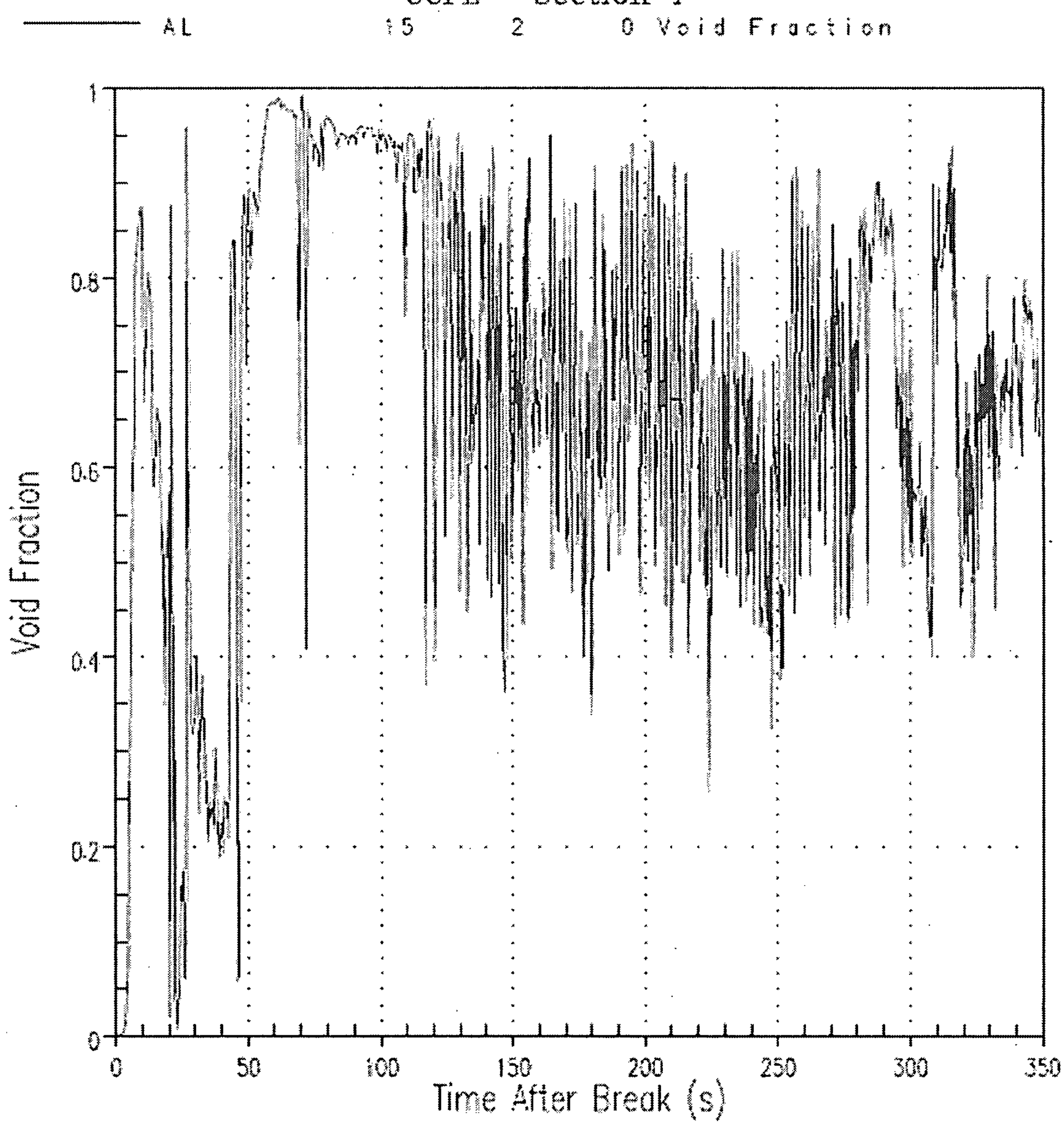
PTN Limiting PCT Case

Figure b39: Void Fraction from Intact Loop Downcomer Channel 69
CCFL – Section 4



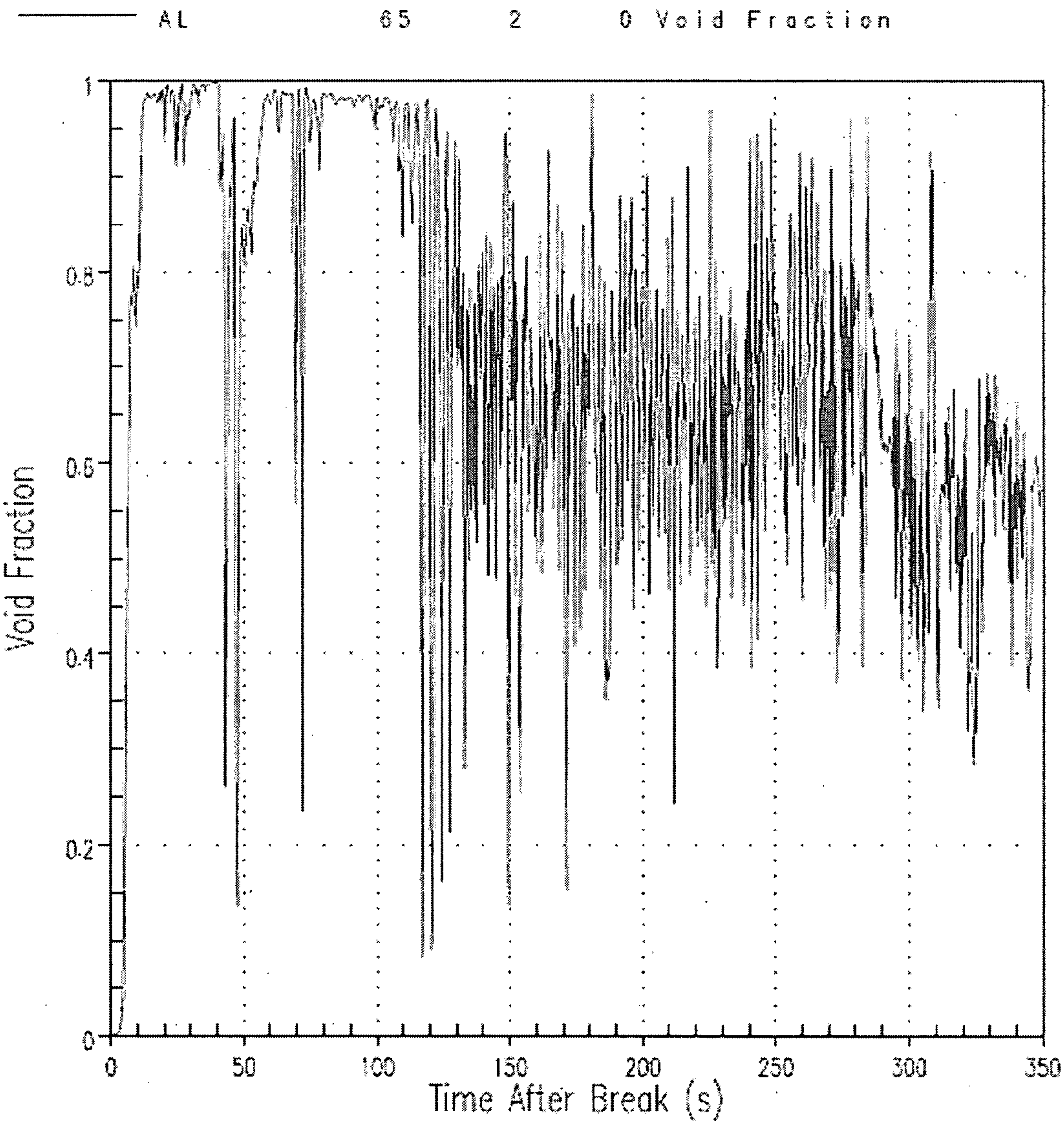
PTN Limiting PCT Case

Figure b40: Void Fraction from Intact Loop Downcomer Channel 15
CCFL – Section 4



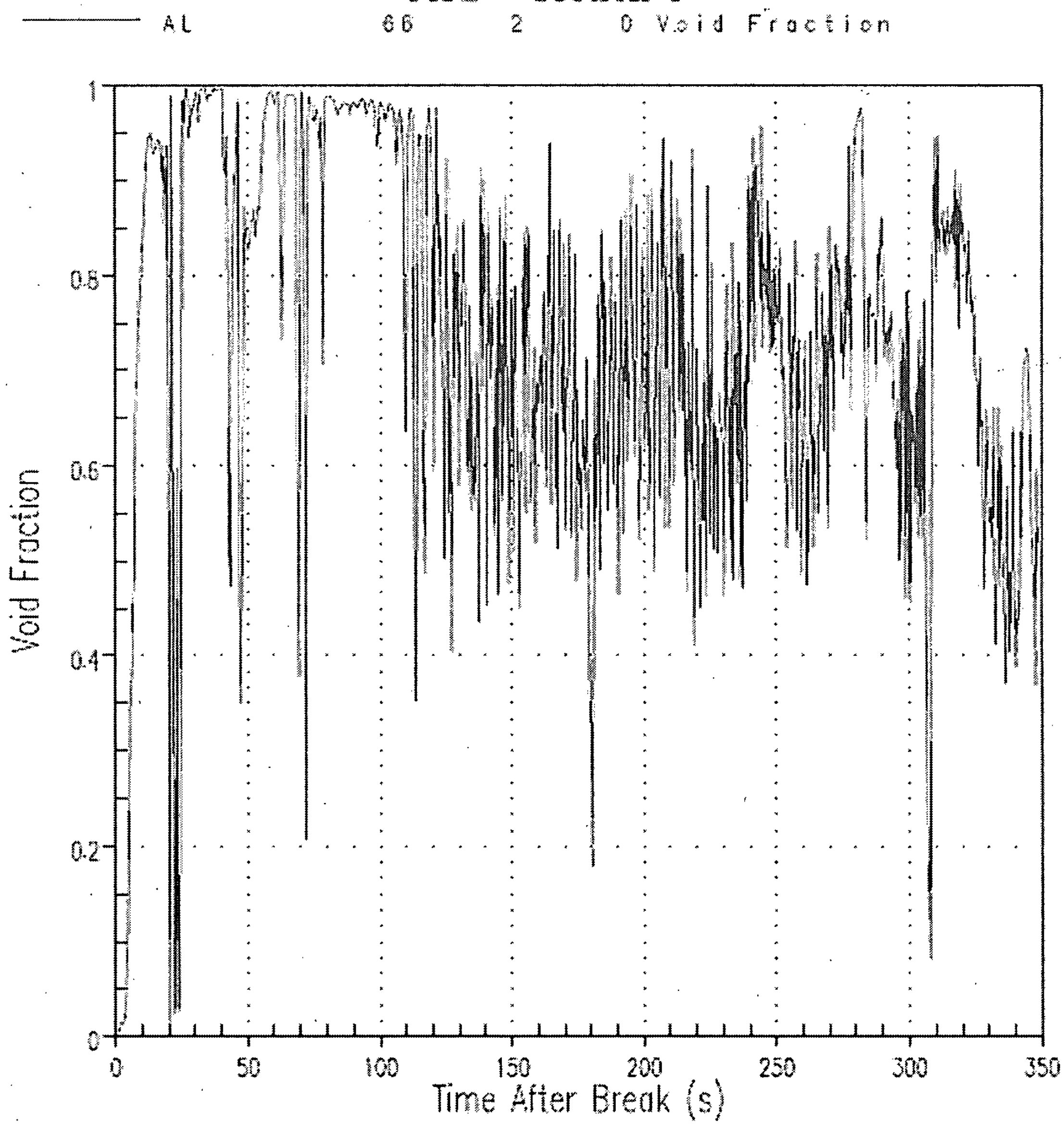
PTN Limiting PCT Case

Figure b41: Void Fraction from Intact Loop Downcomer Channel 65
CCFL – Section 4



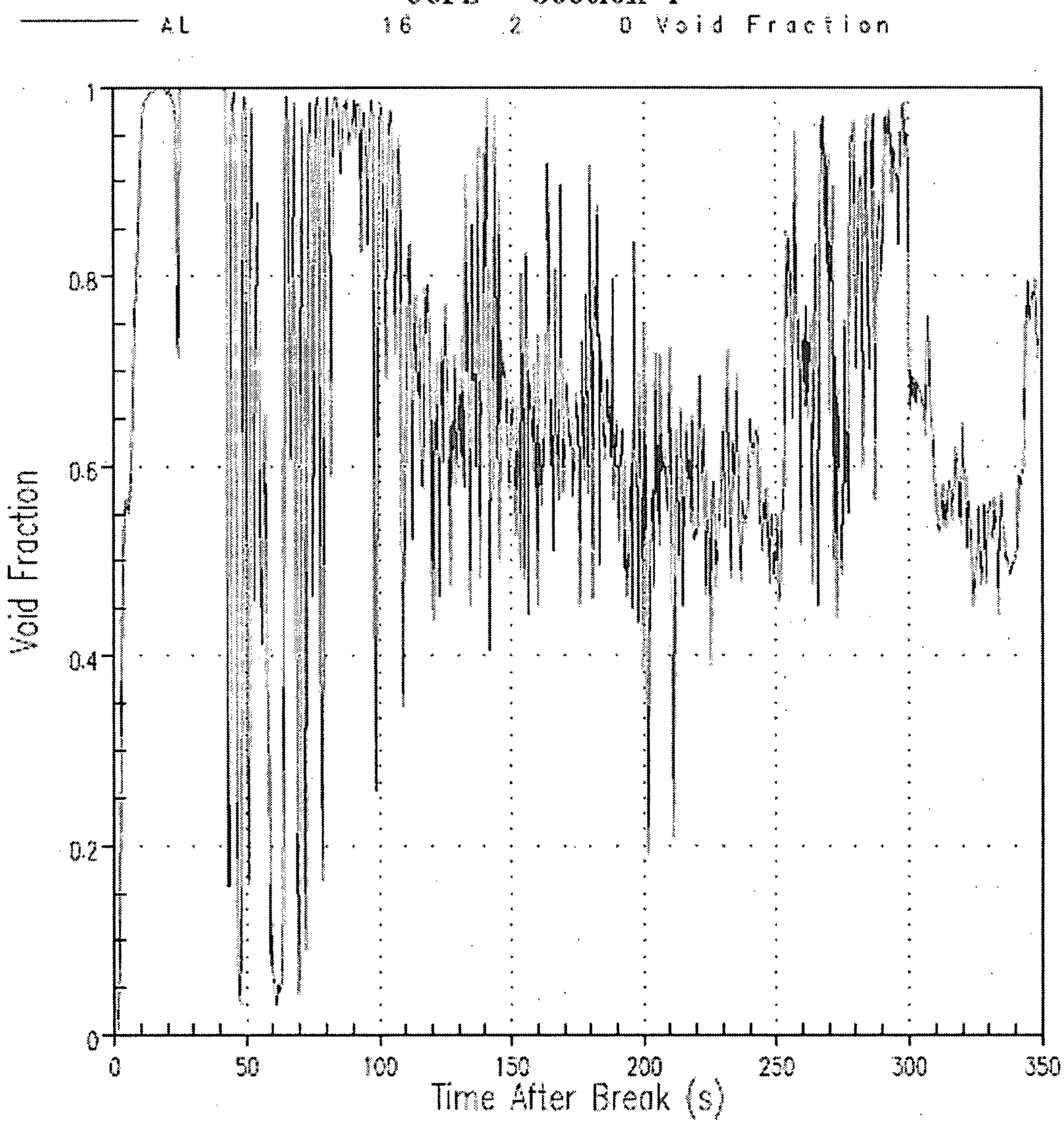
PTN Limiting PCT Case

Figure b42: Void Fraction from Intact Loop Downcomer Channel 66
CCFL - Section 4



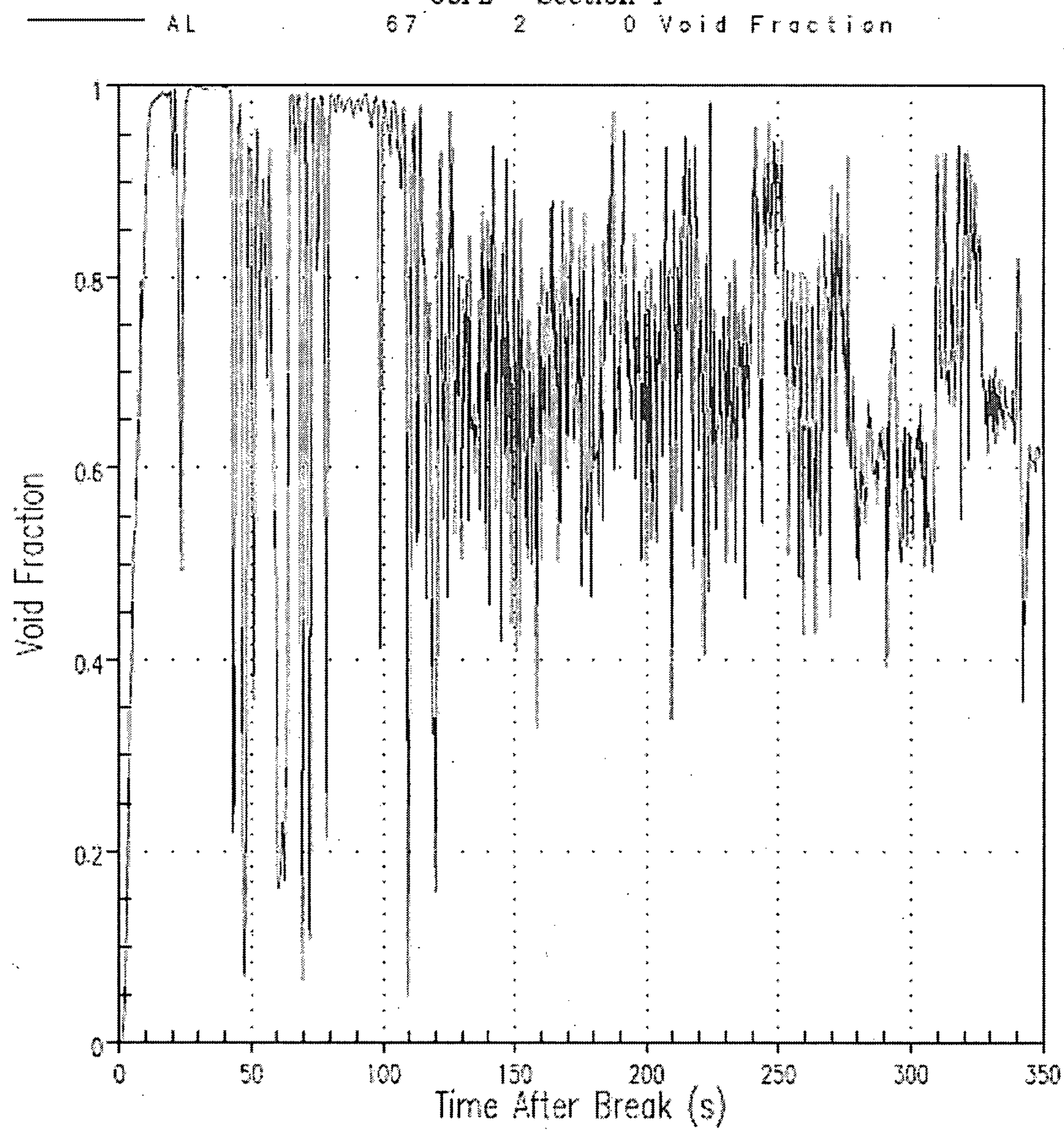
PTN Limiting PCT Case

Figure b43: Void Fraction from Broken Loop Downcomer Channel 16
CCFL – Section 4



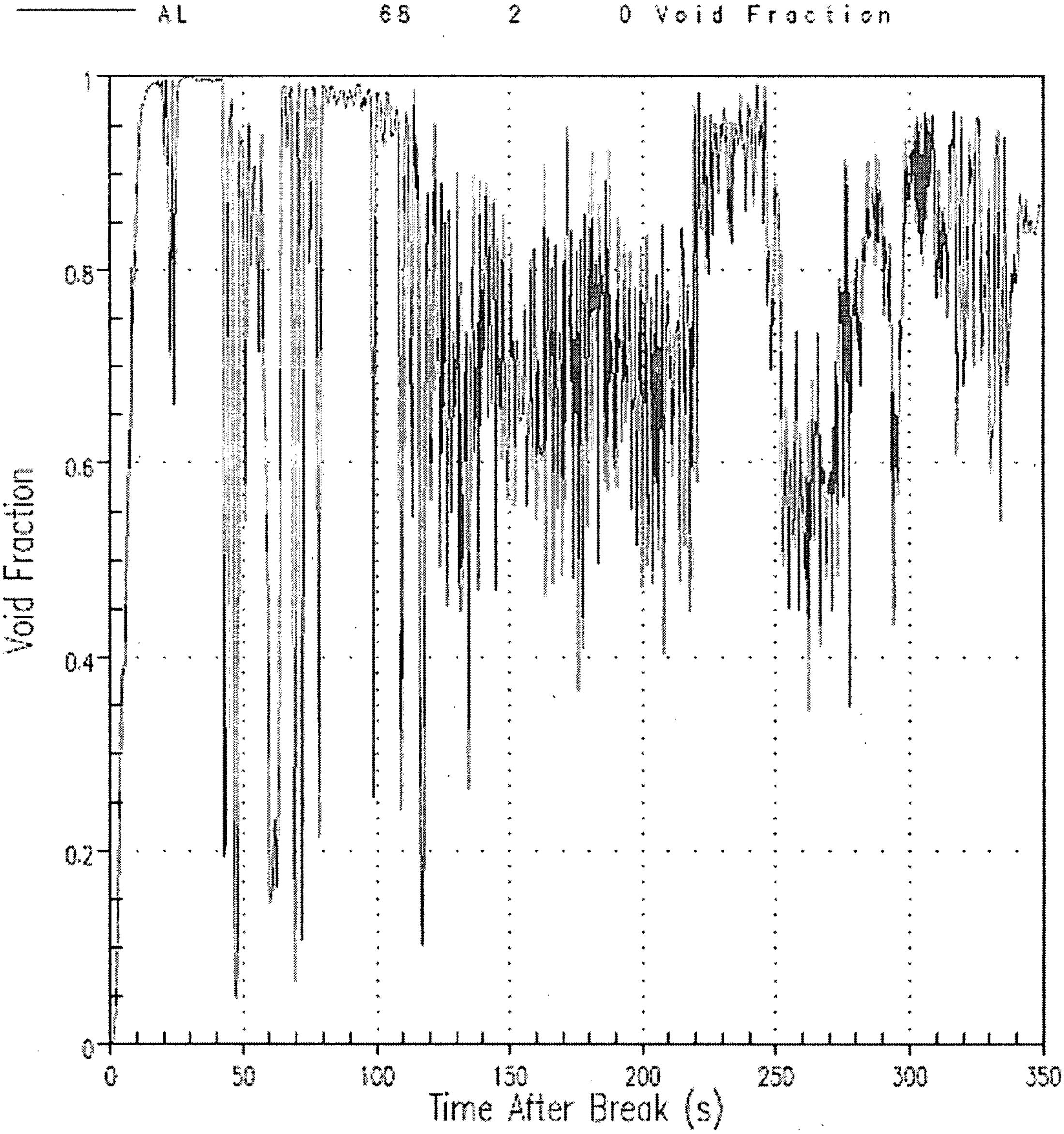
PTN Limiting PCT Case

Figure b44: Void Fraction from Adjacen to Broken Loop Downcomer Channel 67
CCFL – Section 4



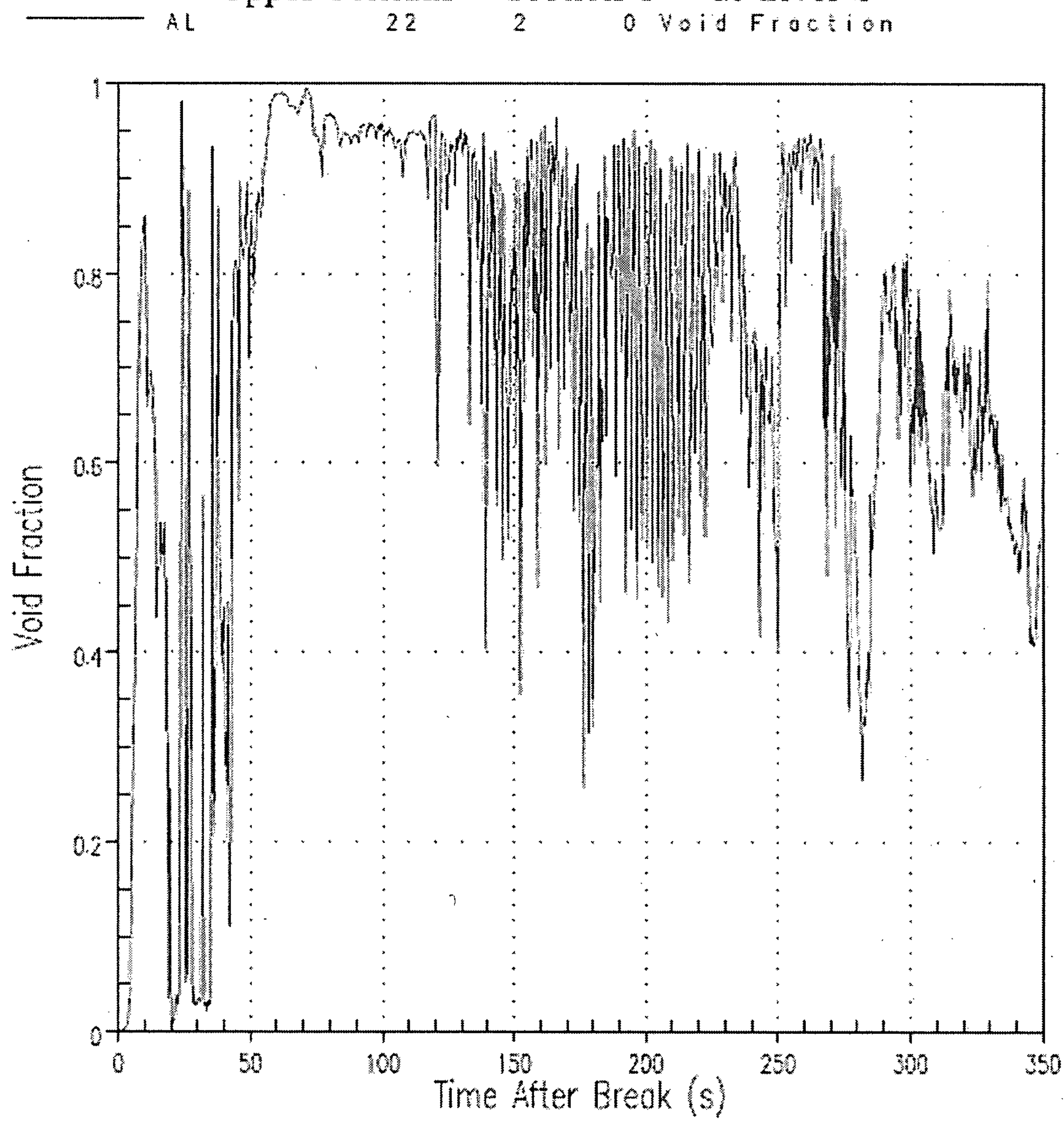
PTN Limiting PCT Case

Figure b45: Void Fraction from Adjacen to Broken Loop Downcomer Channel 68
CCFL – Section 4



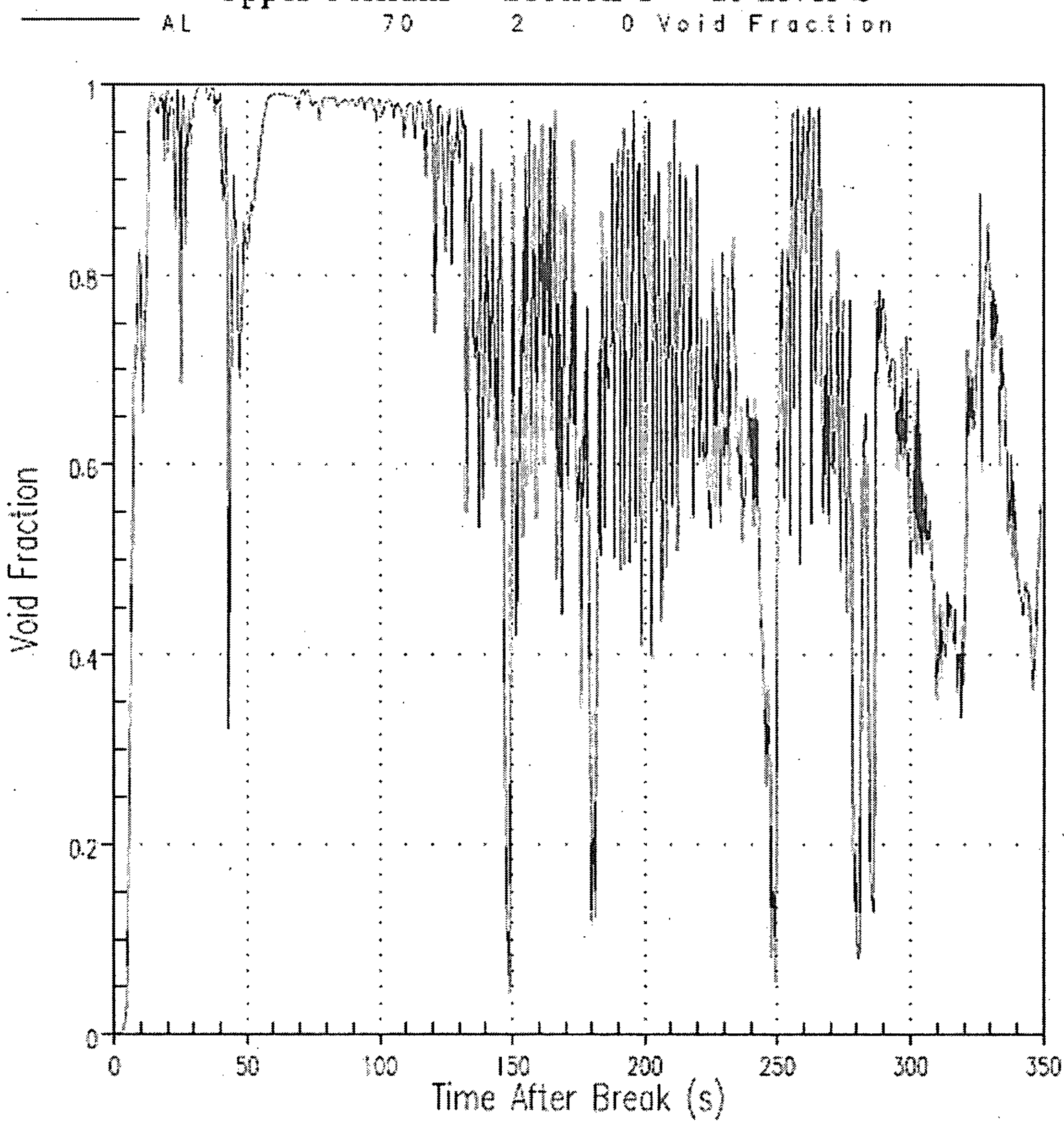
PTN Limiting PCT Case

Figure b46: Void Fraction from Intact Loop Downcomer Channel 22
Upper Plenum – Section 5 – at Level 2



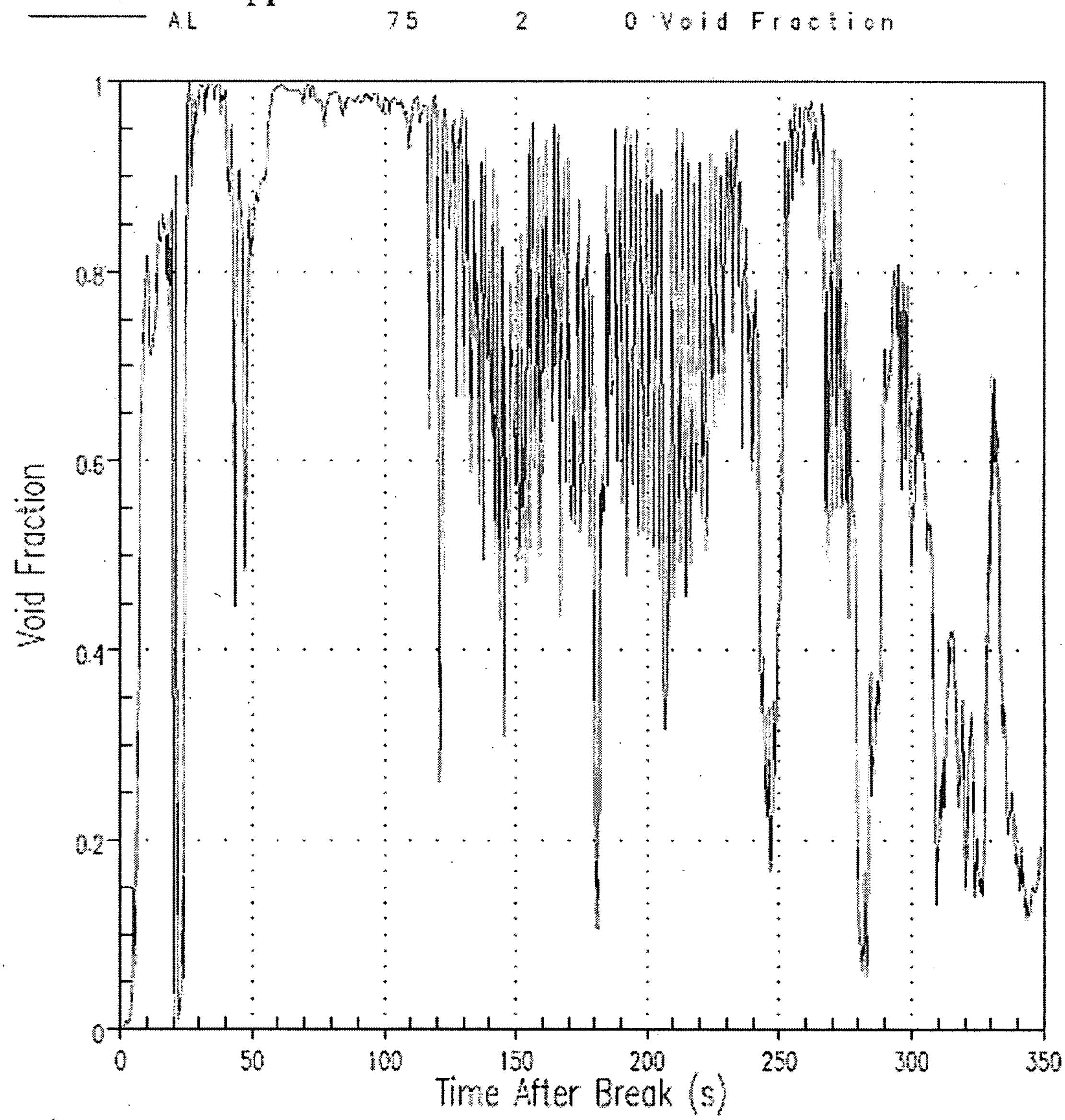
PTN Limiting PCT Case

Figure b47: Void Fraction from Intact Loop Downcomer Channel 70
Upper Plenum - Section 5 - at Level 2



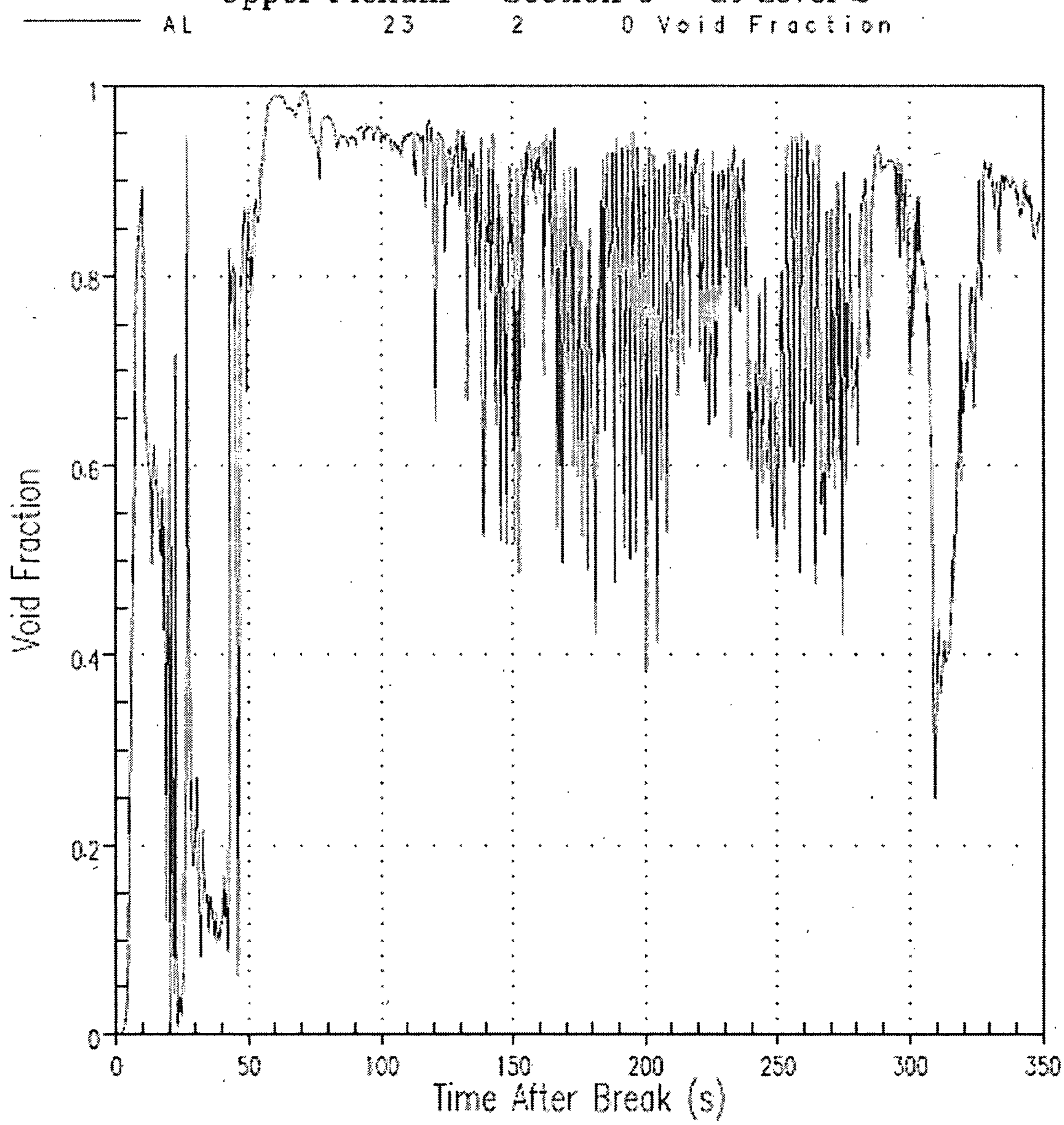
PTN Limiting PCT Case

Figure b48: Void Fraction from Intact Loop Downcomer Channel 75
Upper Plenum – Section 5 – at Level 2



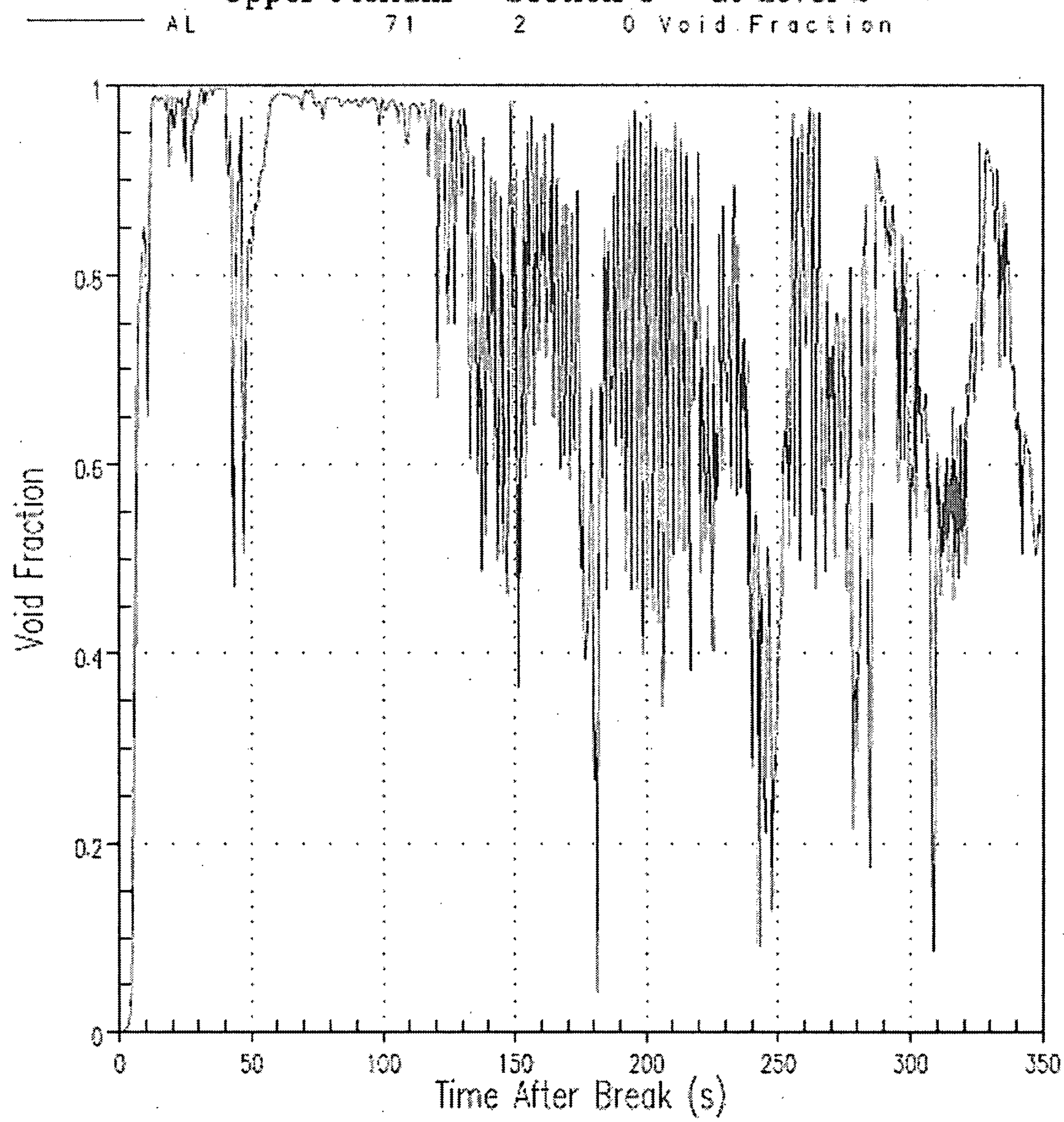
PTN Limiting PCT Case

Figure b49: Void Fraction from Intact Loop Downcomer Channel 23 Upper Plenum – Section 5 – at Level 2



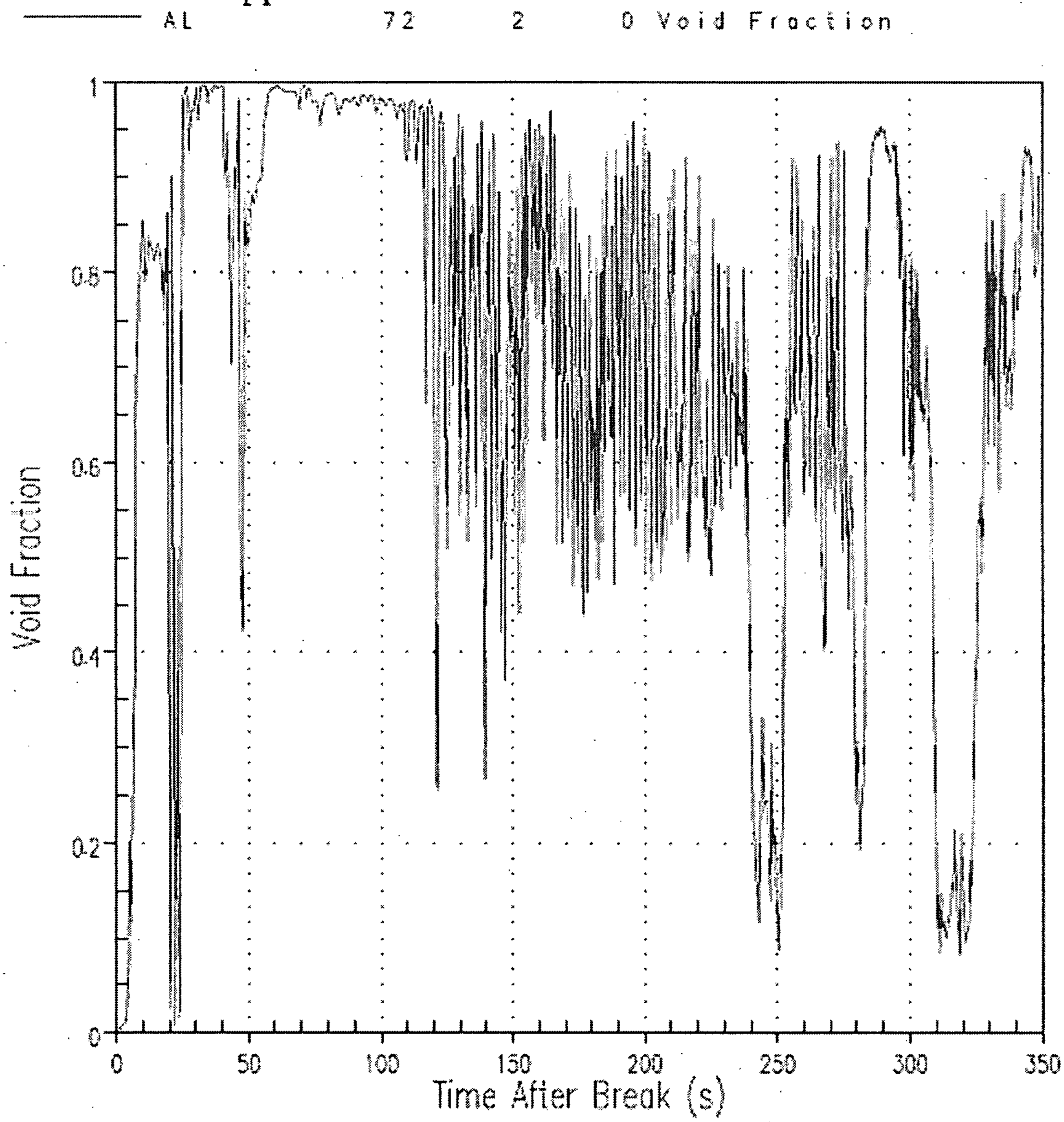
PTN Limiting PCT Case

Figure b50: Void Fraction from Intact Loop Downcomer Channel 71
Upper Plenum – Section 5 – at Level 2



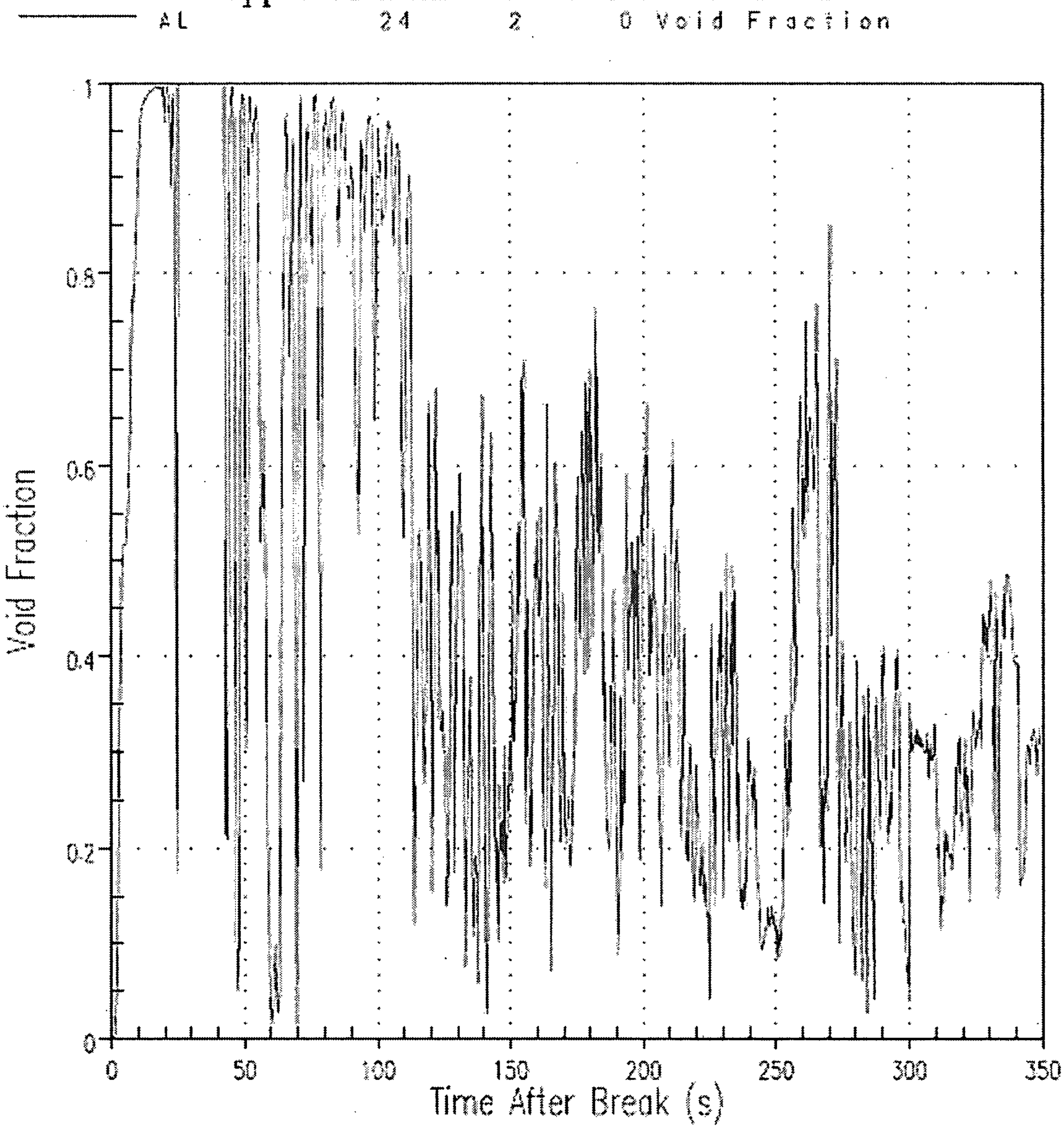
PTN Limiting PCT Case

Figure b51: Void Fraction from Intact Loop Downcomer Channel 72 Upper Plenum – Section 5 – at Level 2



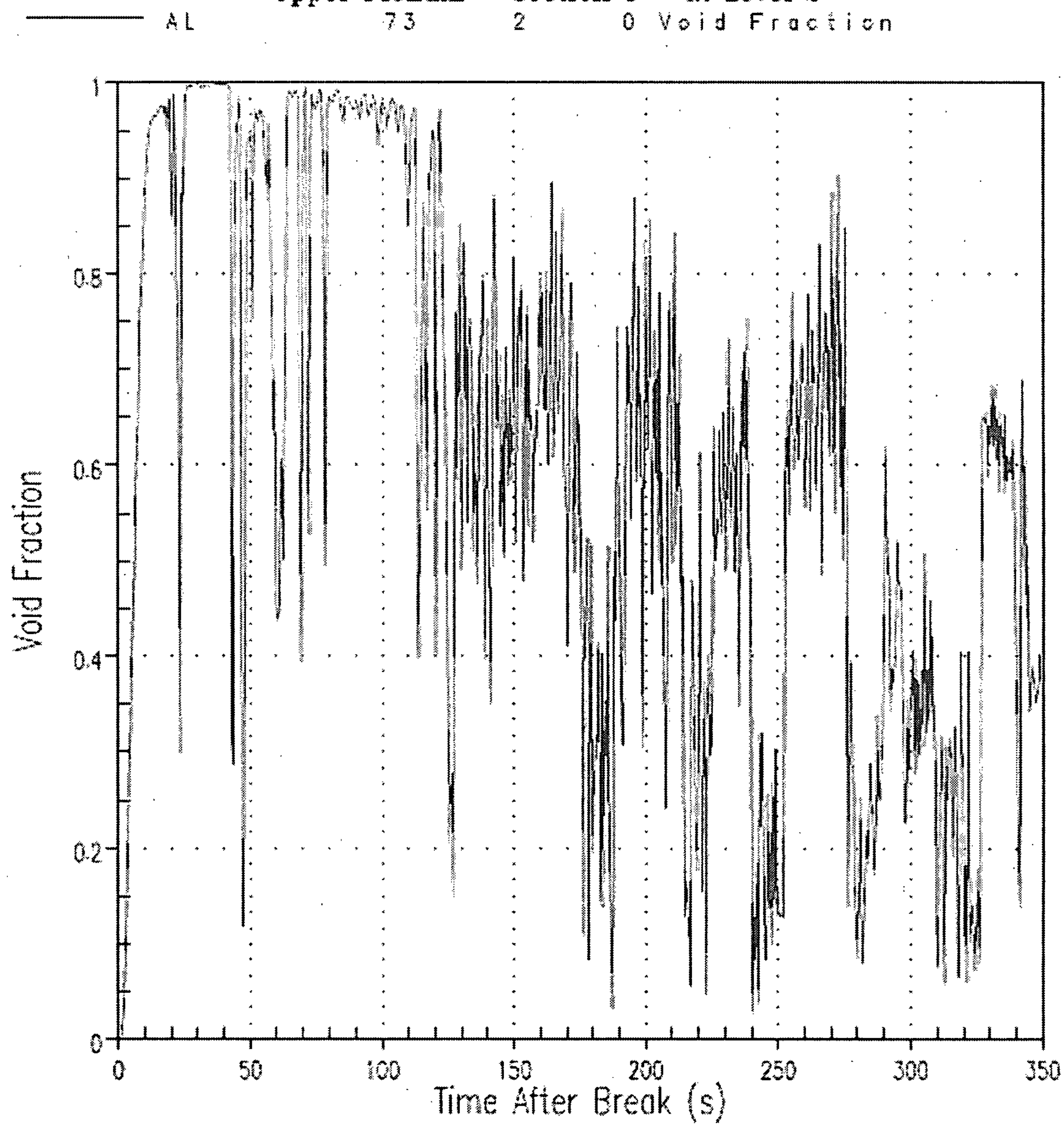
PTN Limiting PCT Case

Figure b52: Void Fraction from Broken Loop Downcomer Channel 24
Upper Plenum - Section 5 - at Level 2



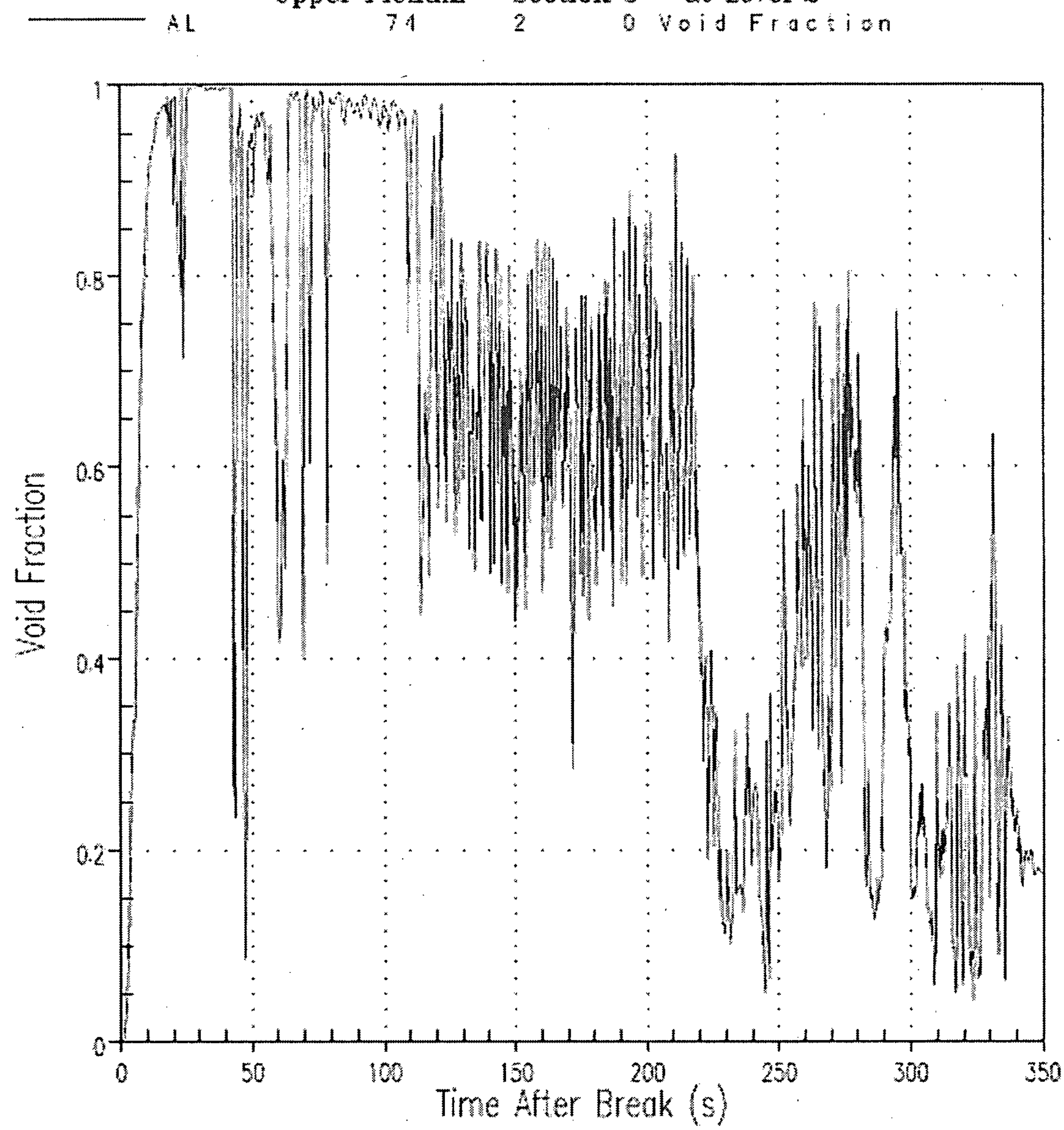
PTN Limiting PCT Case

Figure b53: Void Fraction from Adjacen to Broken Loop Downcomer Channel 73 Upper Plenum – Section 5 – at Level 2



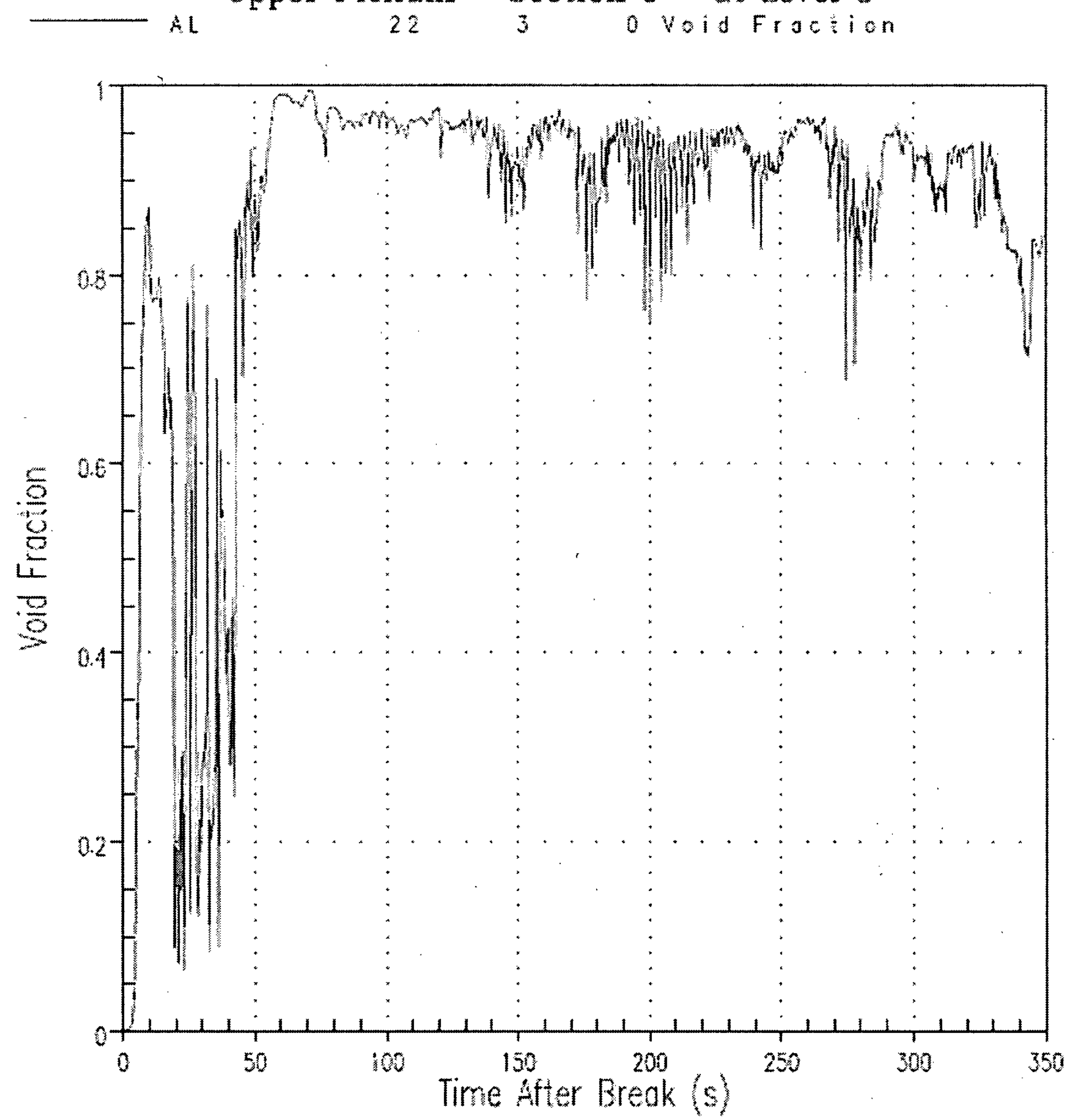
PTN Limiting PCT Case

Figure b54: Void Fraction from Adjacen to Broken Loop Downcomer Channel 74
Upper Plenum – Section 5 – at Level 2



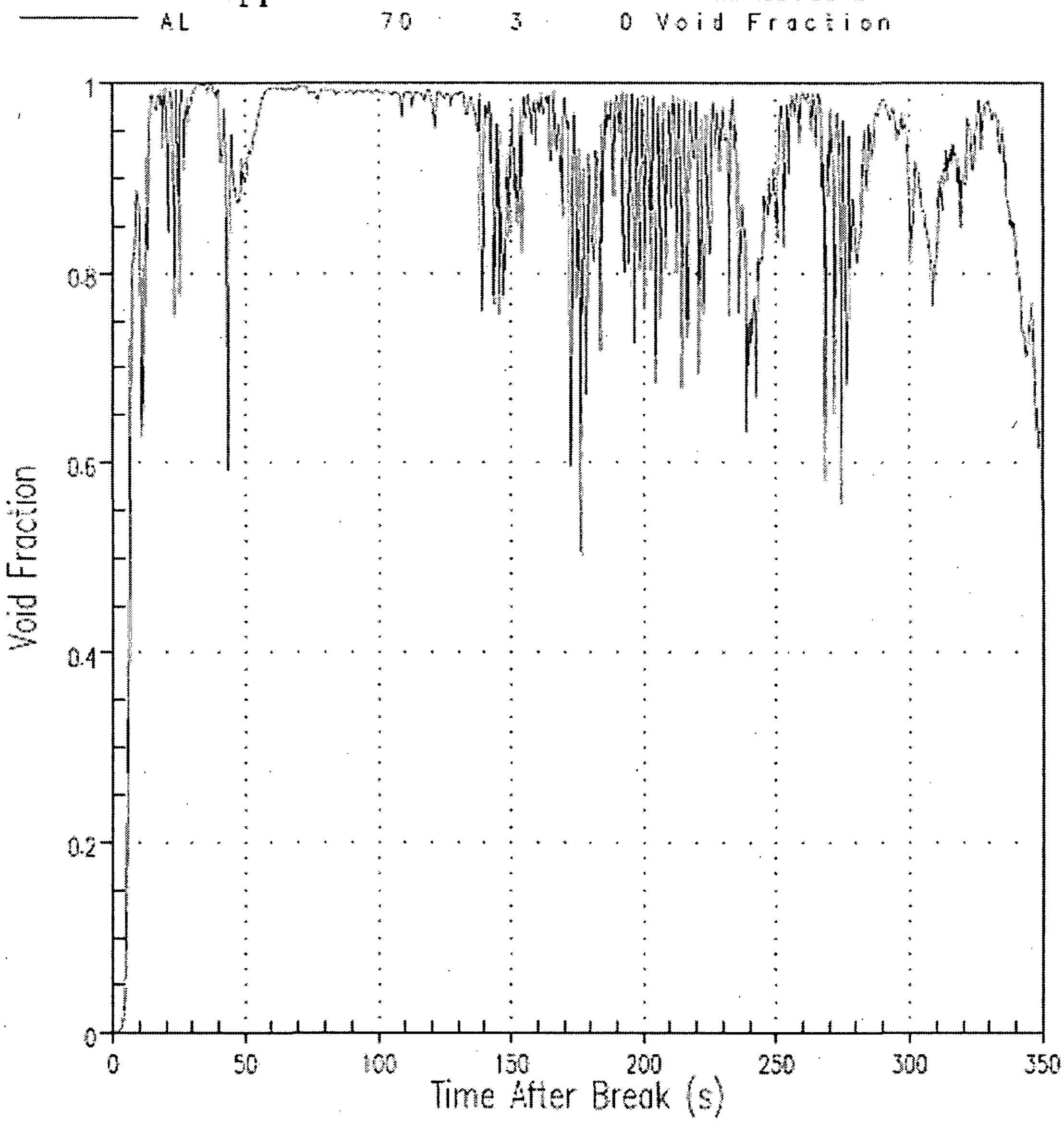
PTN Limiting PCT Case

Figure b55: Void Fraction from Intact Loop Downcomer Channel 22
Upper Plenum – Section 5 – at Level 3



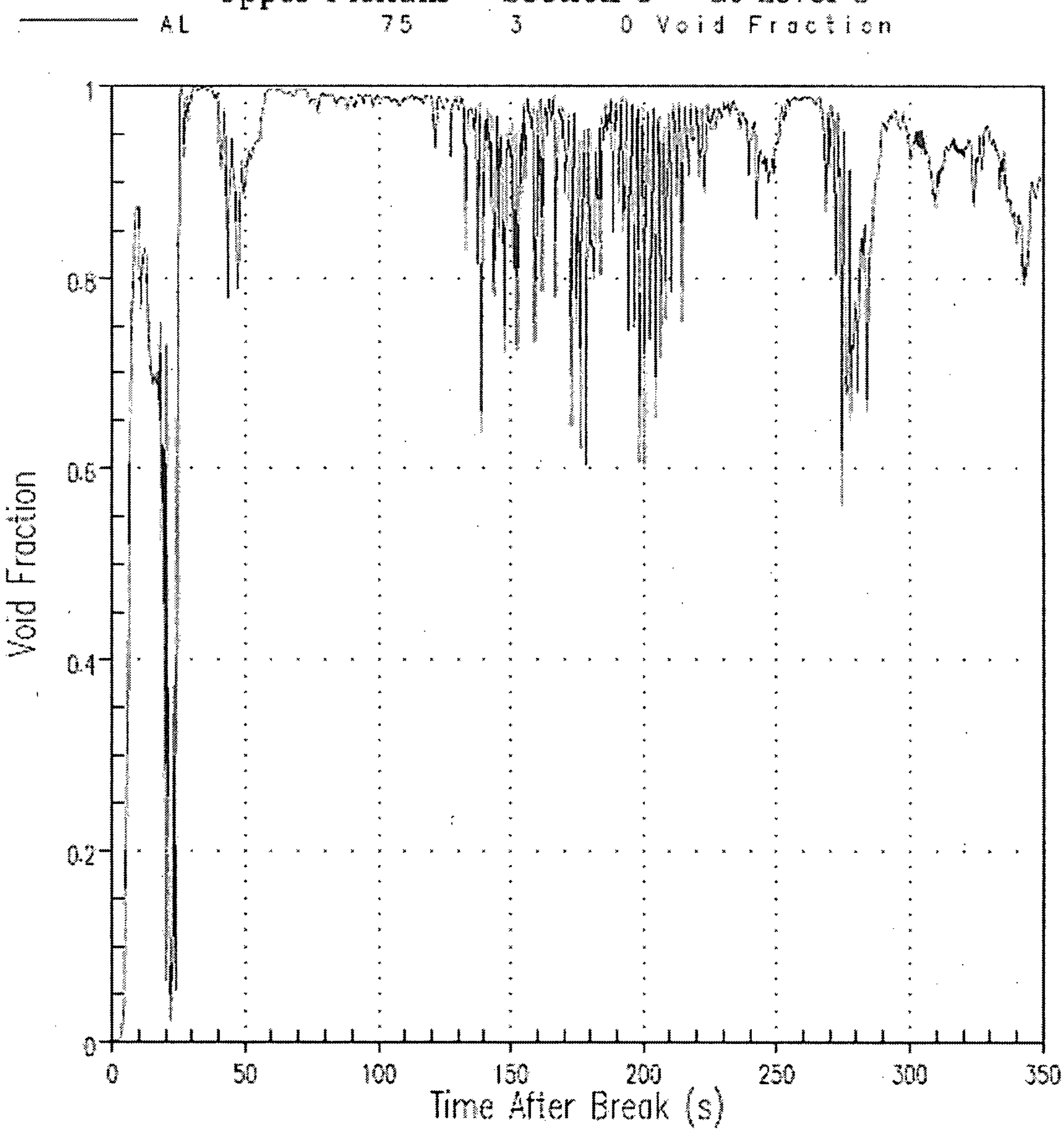
PTN Limiting PCT Case

Figure b56: Void Fraction from Intact Loop Downcomer Channel 70
Upper Plenum – Section 5 – at Level 3



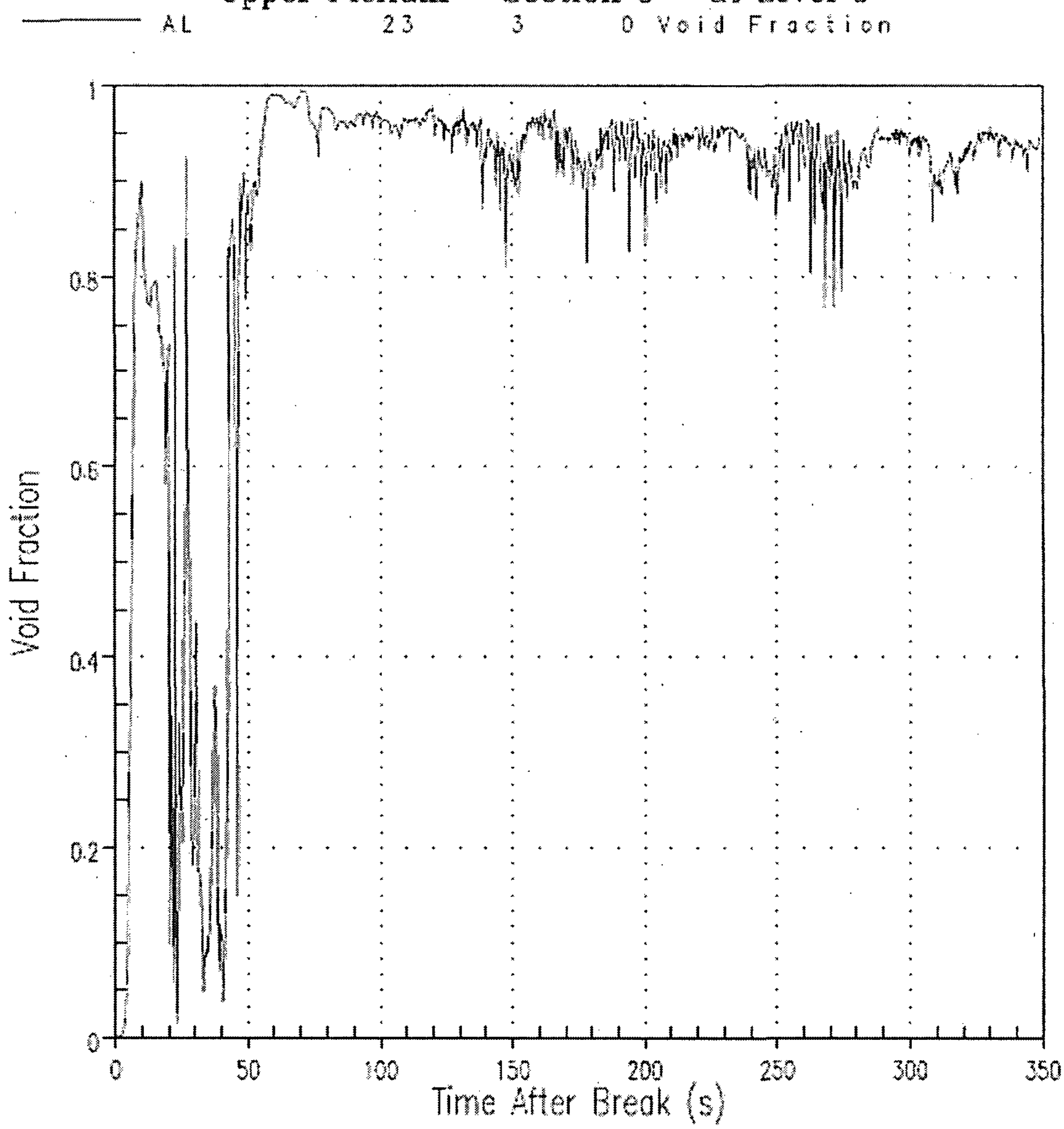
PTN Limiting PCT Case

Figure b57: Void Fraction from Intact Loop Downcomer Channel 75
Upper Plenum – Section 5 – at Level 3



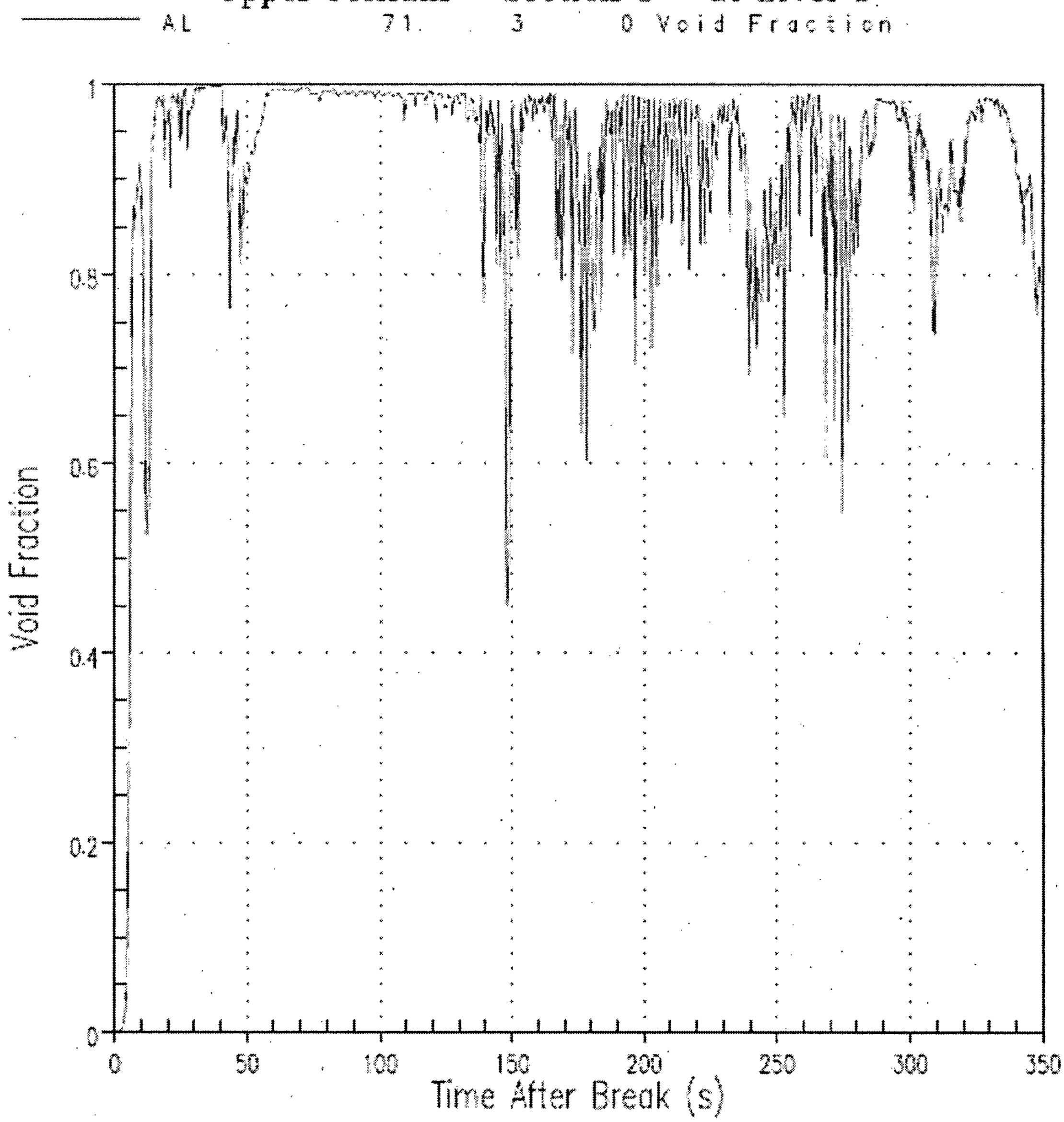
PTN Limiting PCT Case

Figure b58: Void Fraction from Intact Loop Downcomer Channel 23 Upper Plenum – Section 5 – at Level 3



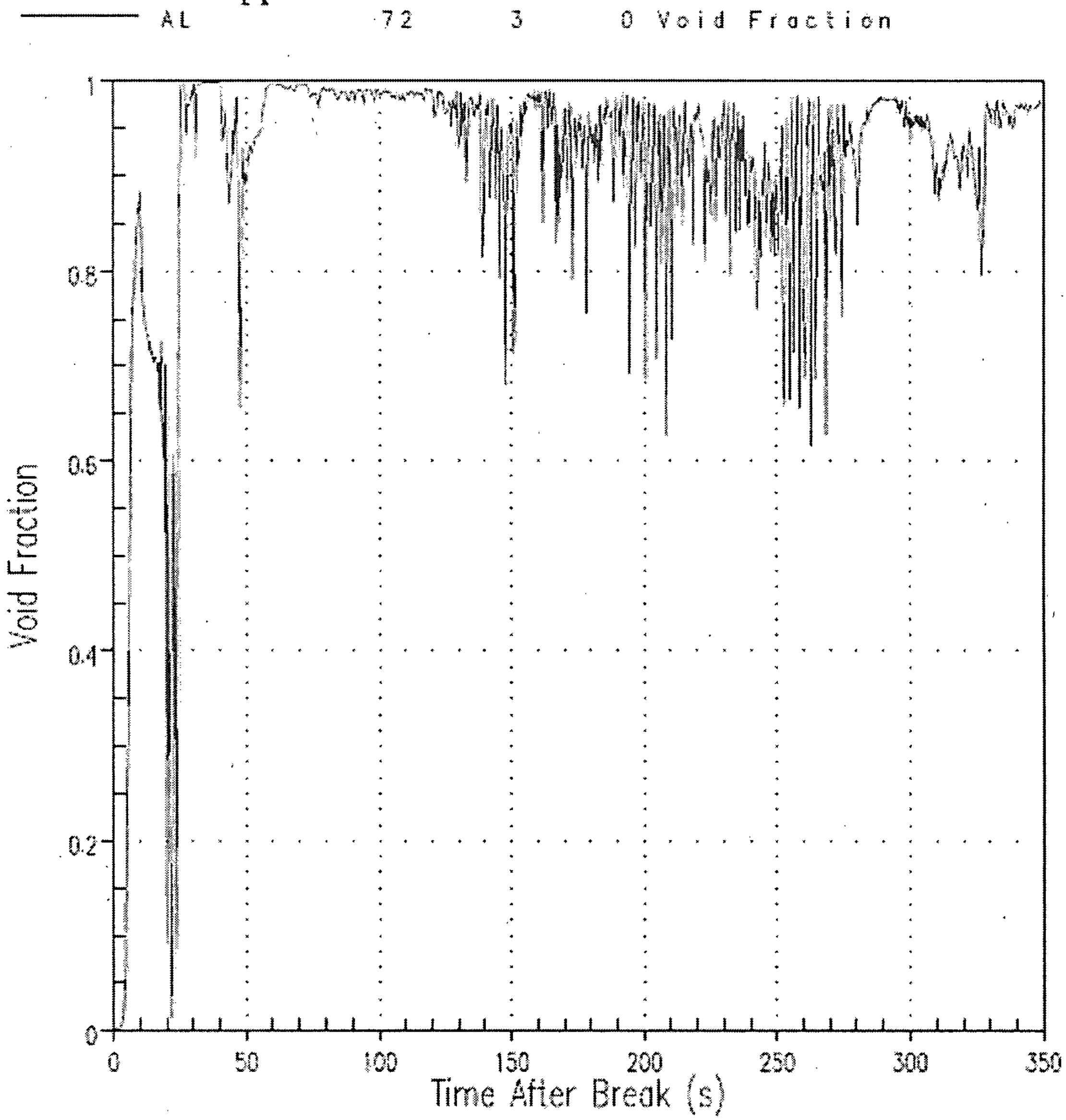
PTN Limiting PCT Case

Figure b59: Void Fraction from Intact Loop Downcomer Channel 71
Upper Plenum – Section 5 – at Level 3



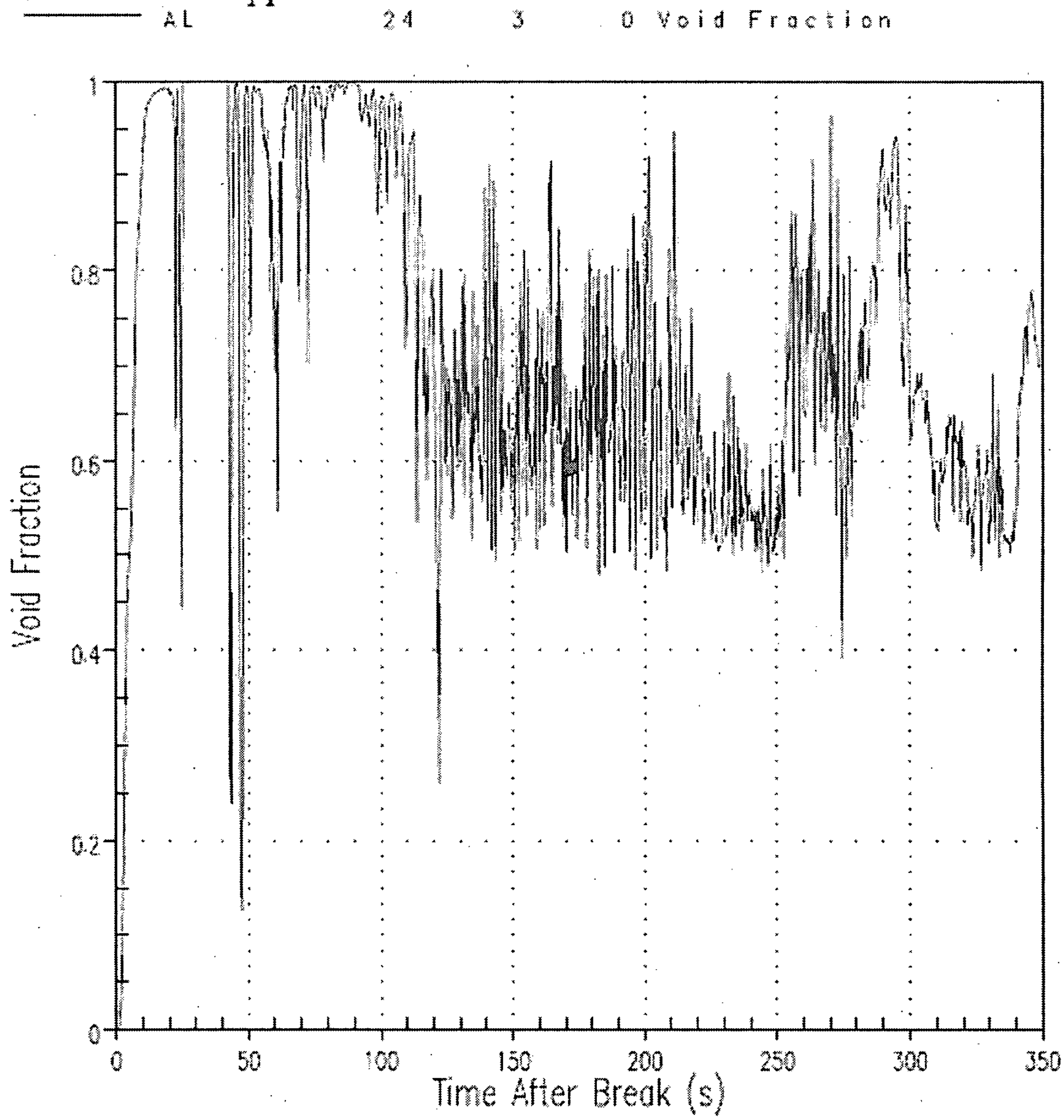
PTN Limiting PCT Case

Figure b60: Void Fraction from Intact Loop Downcomer Channel 72
Upper Plenum – Section 5 – at Level 3



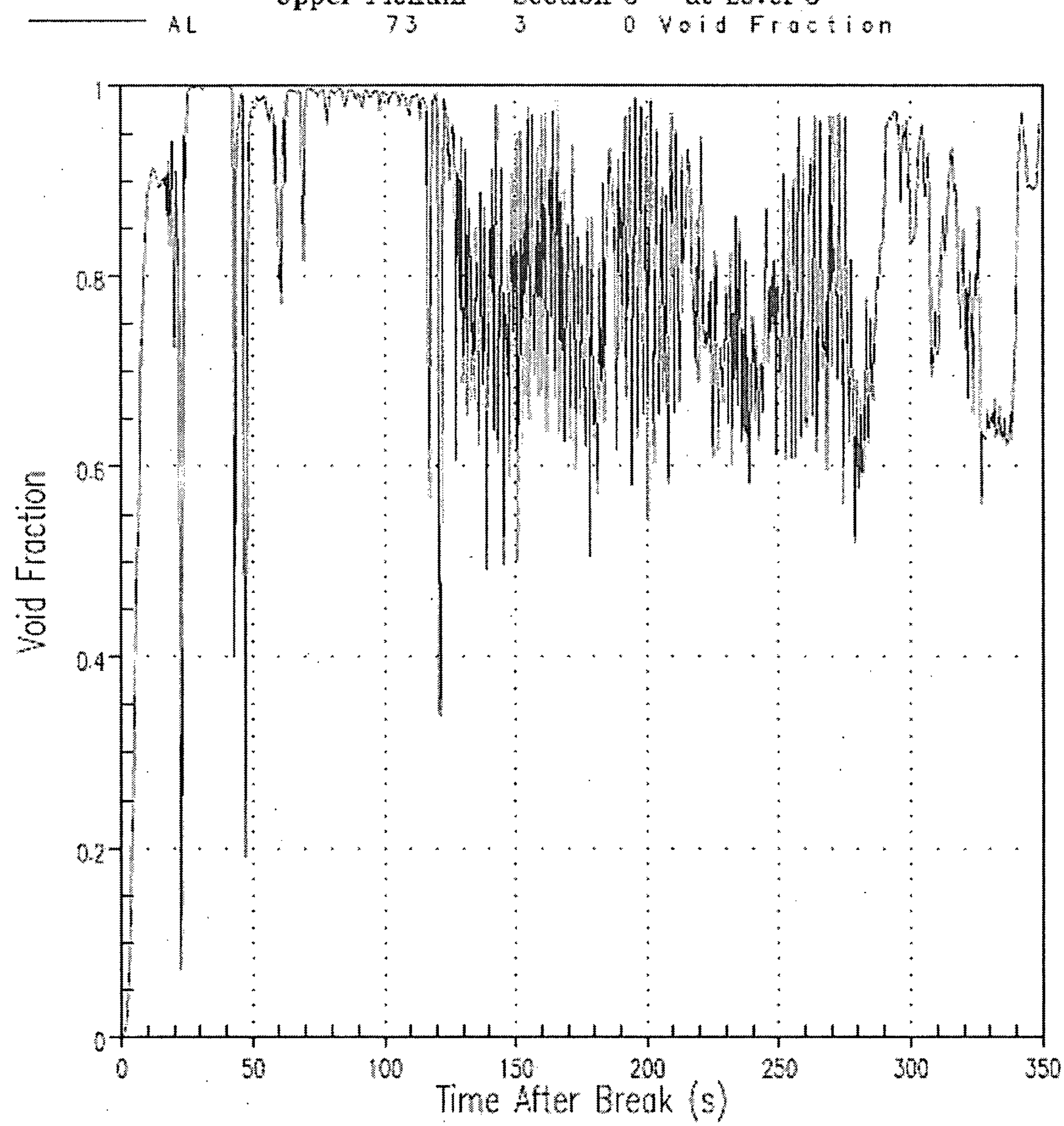
PTN Limiting PCT Case

Figure b61: Void Fraction from Broken Loop Downcomer Channel 24
Upper Plenum - Section 5 - at Level 3



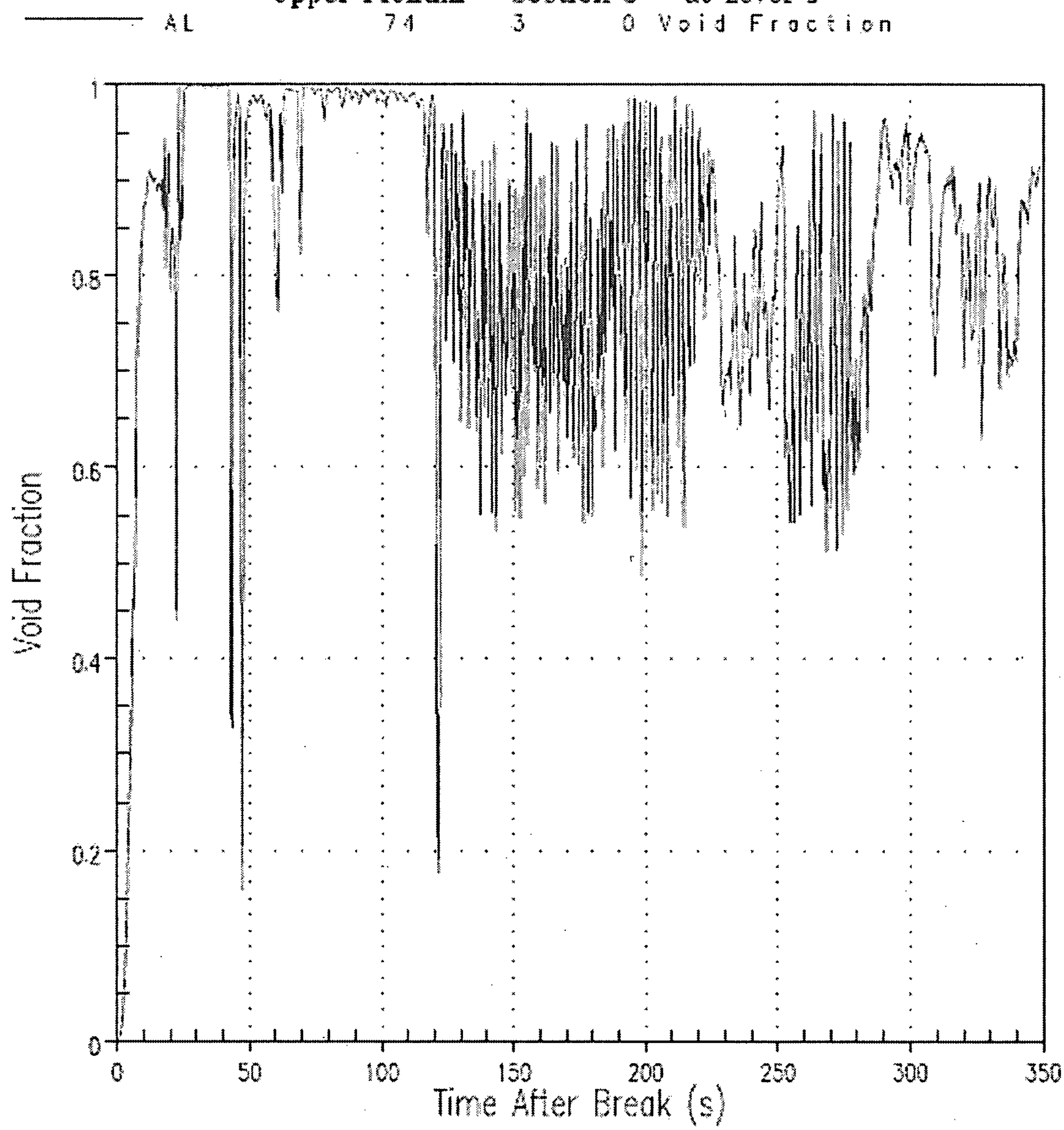
PTN Limiting PCT Case

Figure b62: Void Fraction from Adjacen to Broken Loop Downcomer Channel 73
Upper Plenum – Section 5 – at Level 3



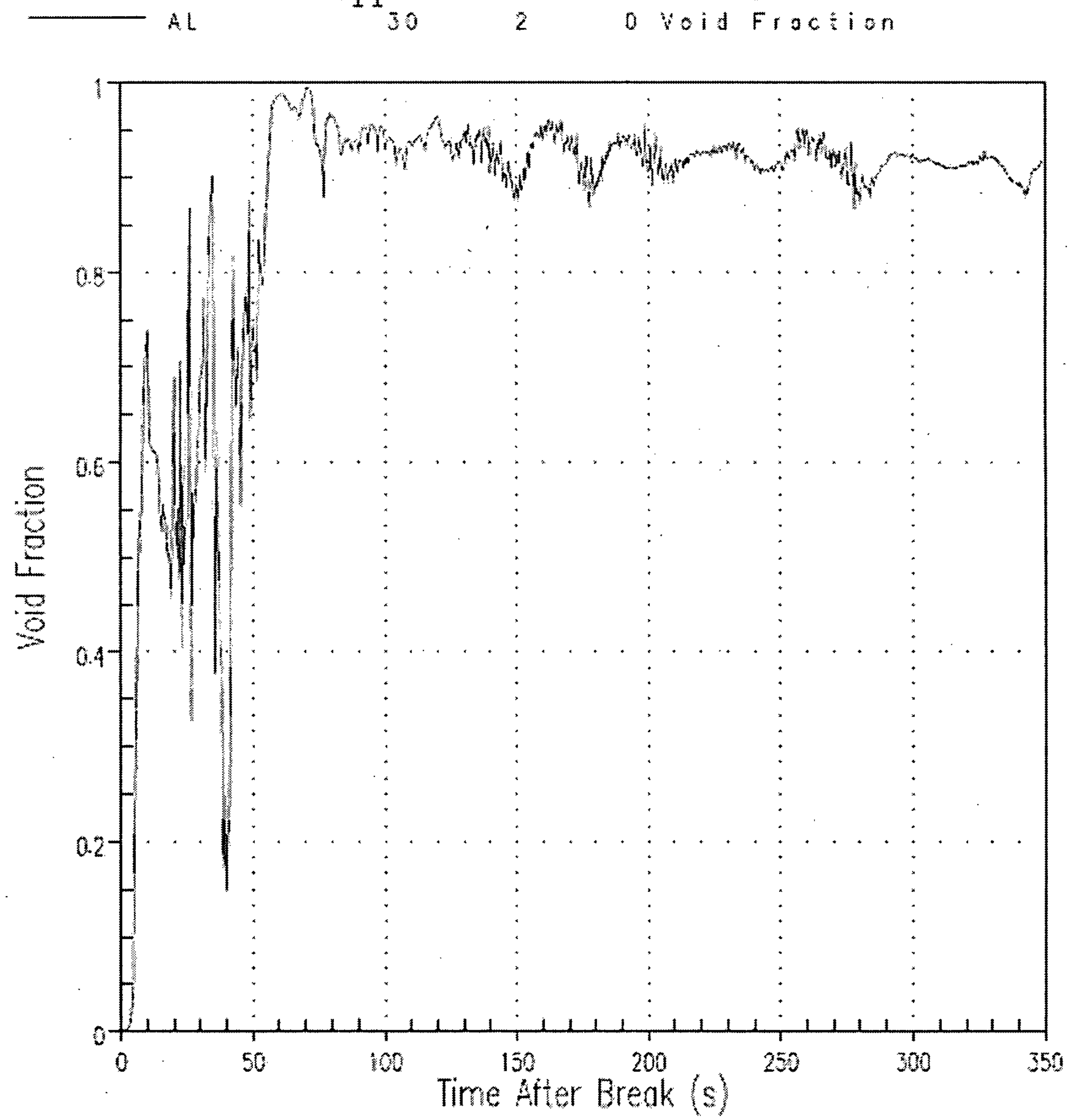
PTN Limiting PCT Case

Figure b63: Void Fraction from Adjacen to Broken Loop Downcomer Channel 74
Upper Plenum – Section 5 – at Level 3



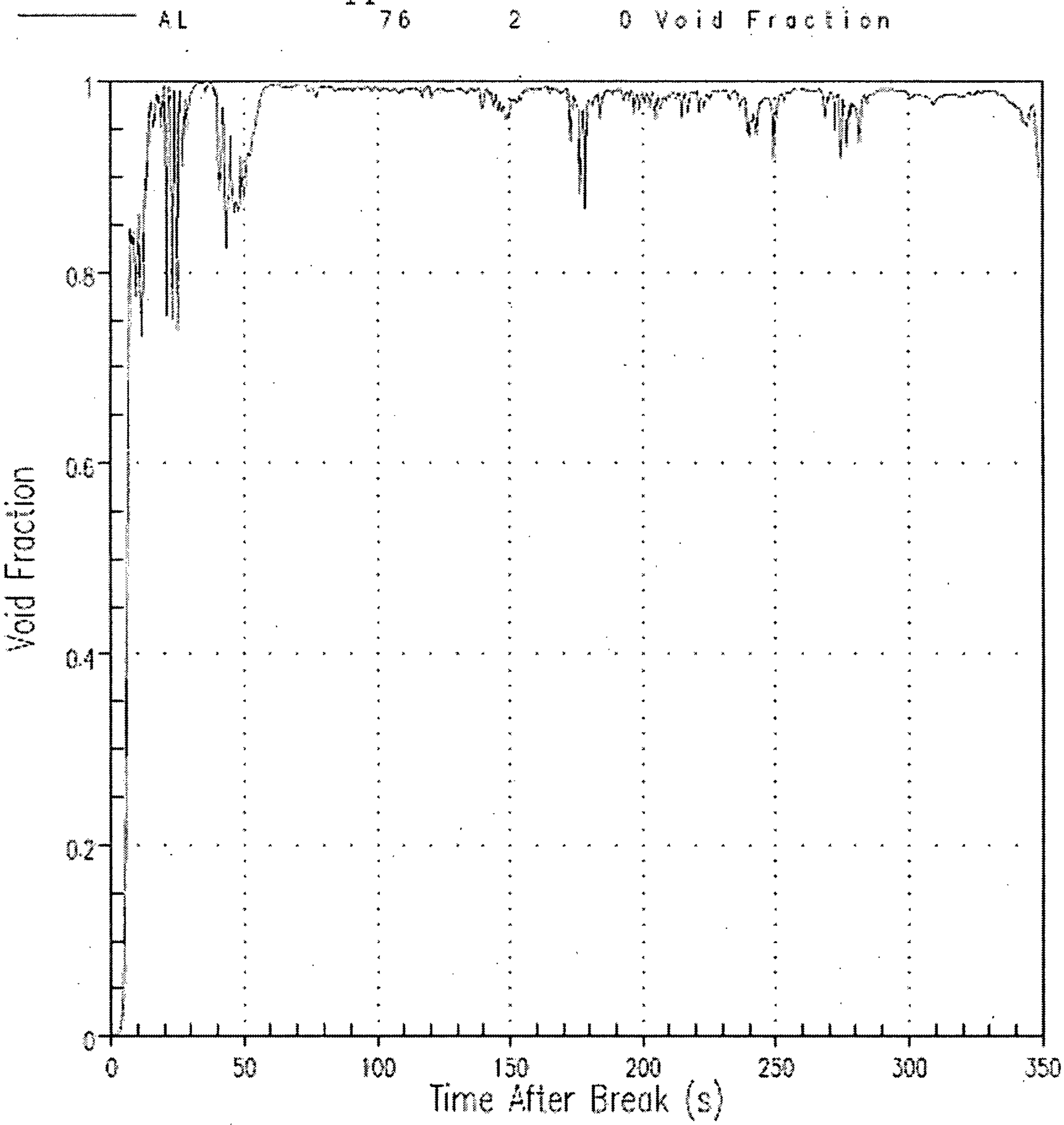
PTN Limiting PCT Case

Figure b64: Void Fraction from Intact Loop Downcomer Channel 30
Upper Plenum - Section 6



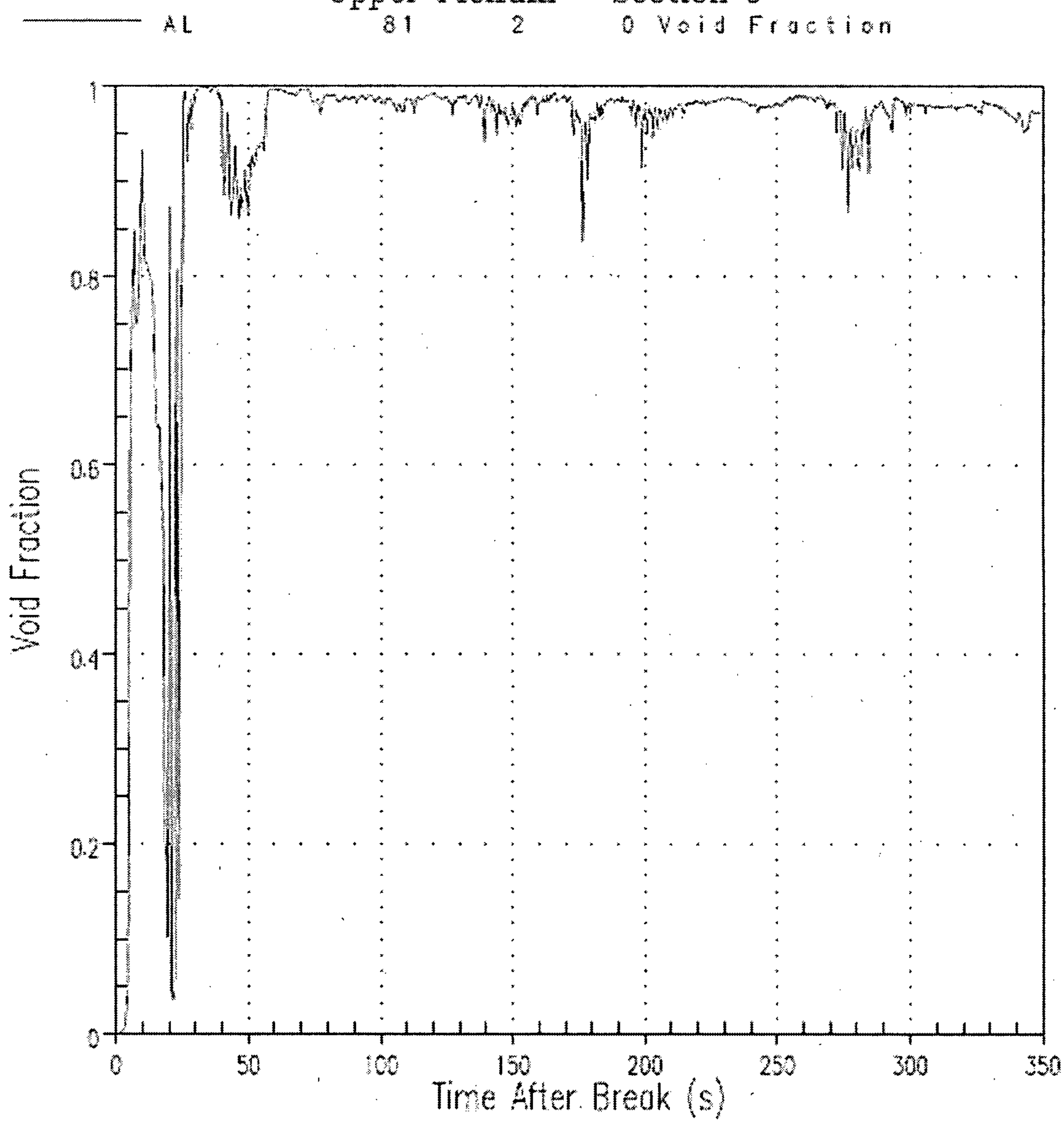
PTN Limiting PCT Case

Figure b65: Void Fraction from Intact Loop Downcomer Channel 76
Upper Plenum – Section 6



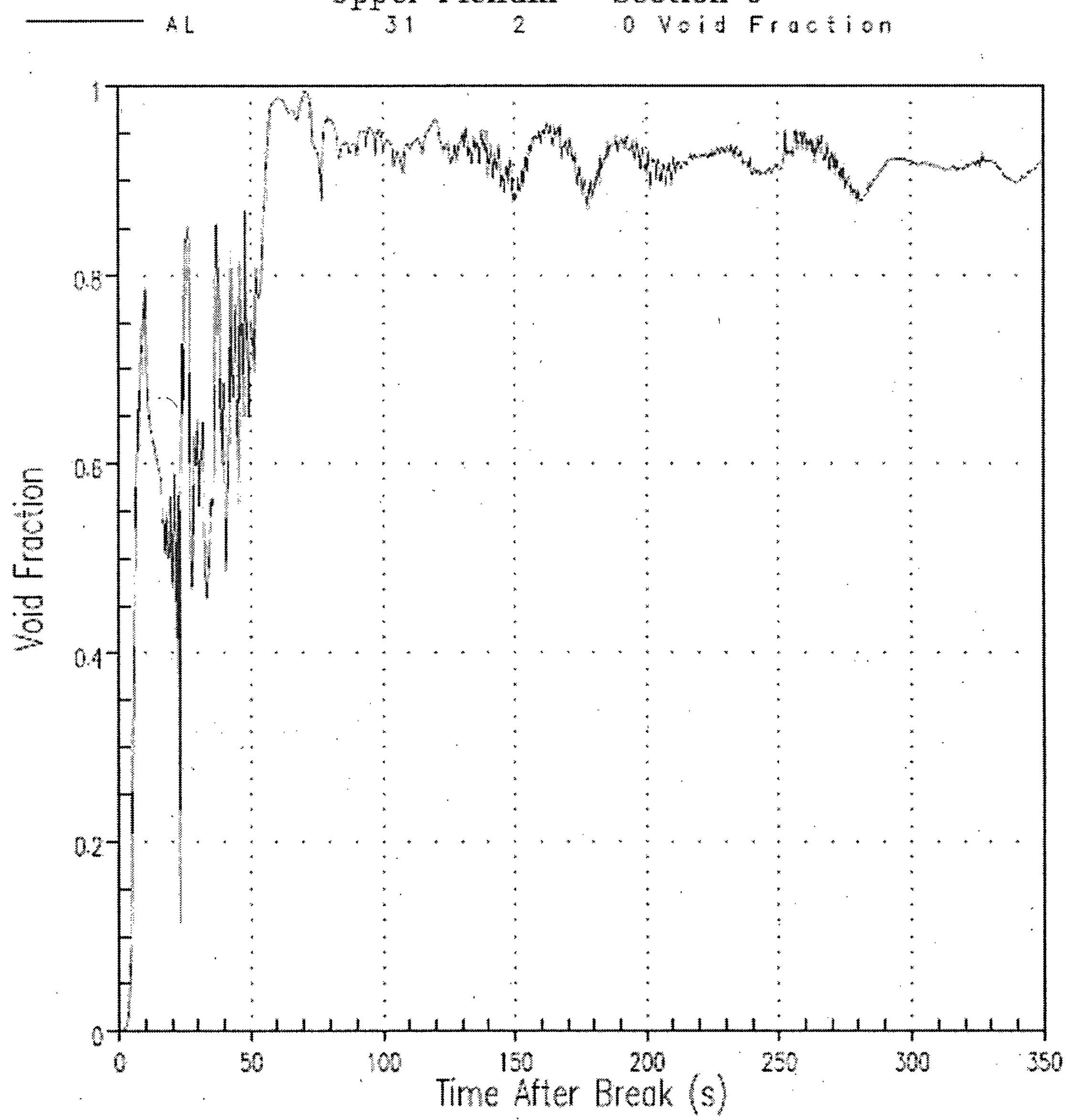
PTN Limiting PCT Case

Figure b66: Void Fraction from Intact Loop Downcomer Channel 81
Upper Plenum – Section 6



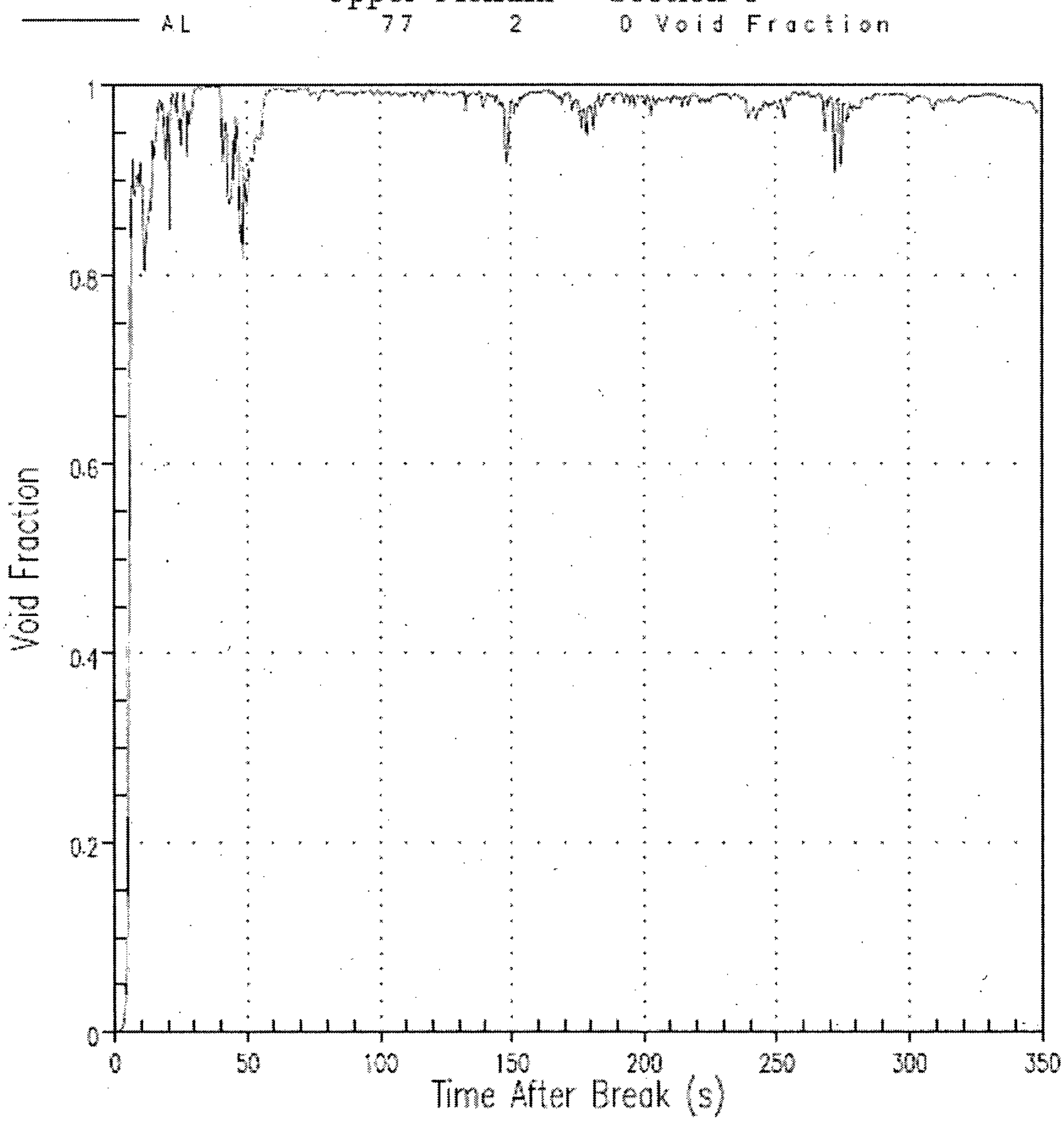
PTN Limiting PCT Case

Figure b67: Void Fraction from Intact Loop Downcomer Channel 31
Upper Plenum – Section 6



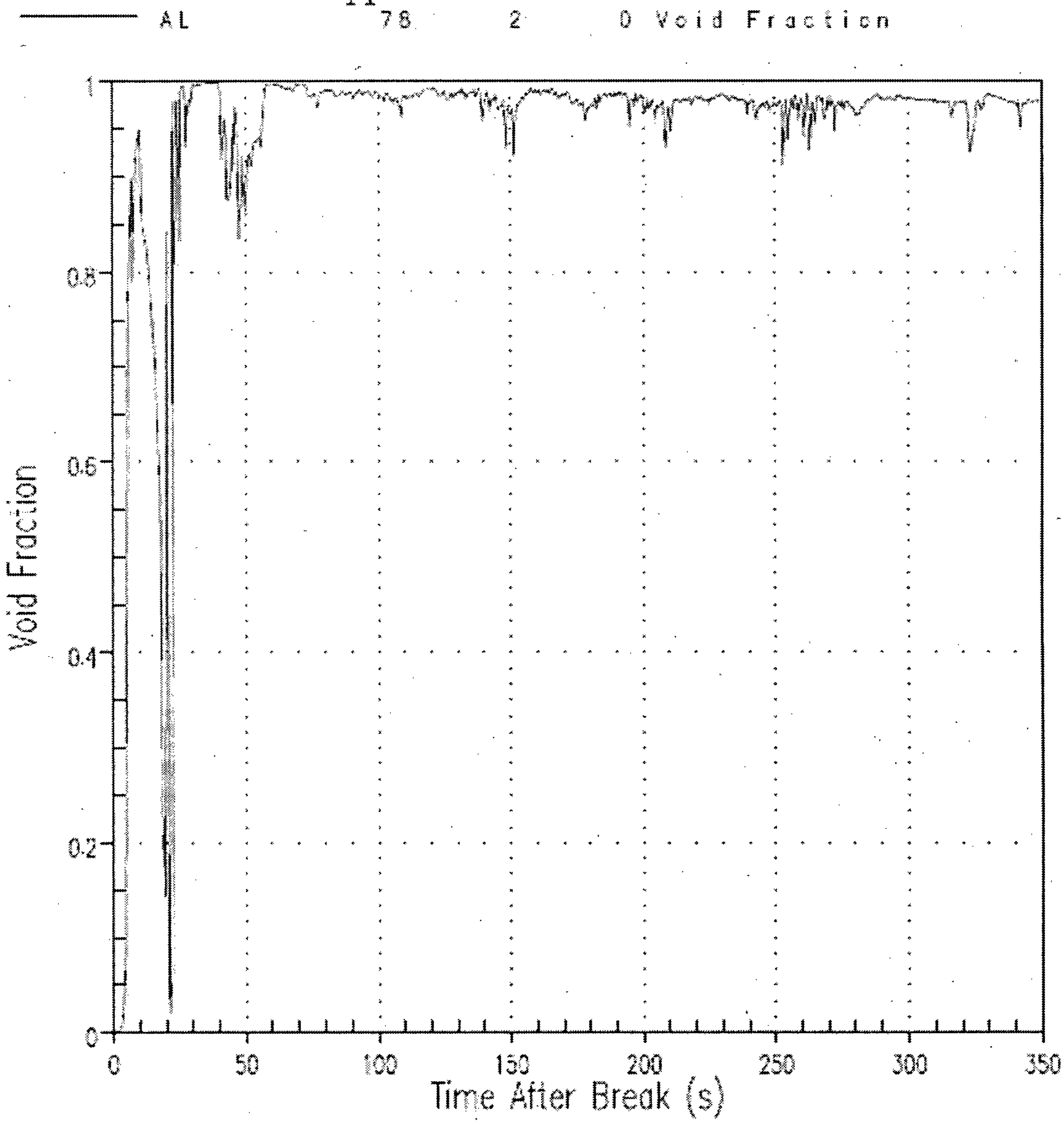
PTN Limiting PCT Case

Figure b68: Void Fraction from Intact Loop Downcomer Channel 77
Upper Plenum – Section 6



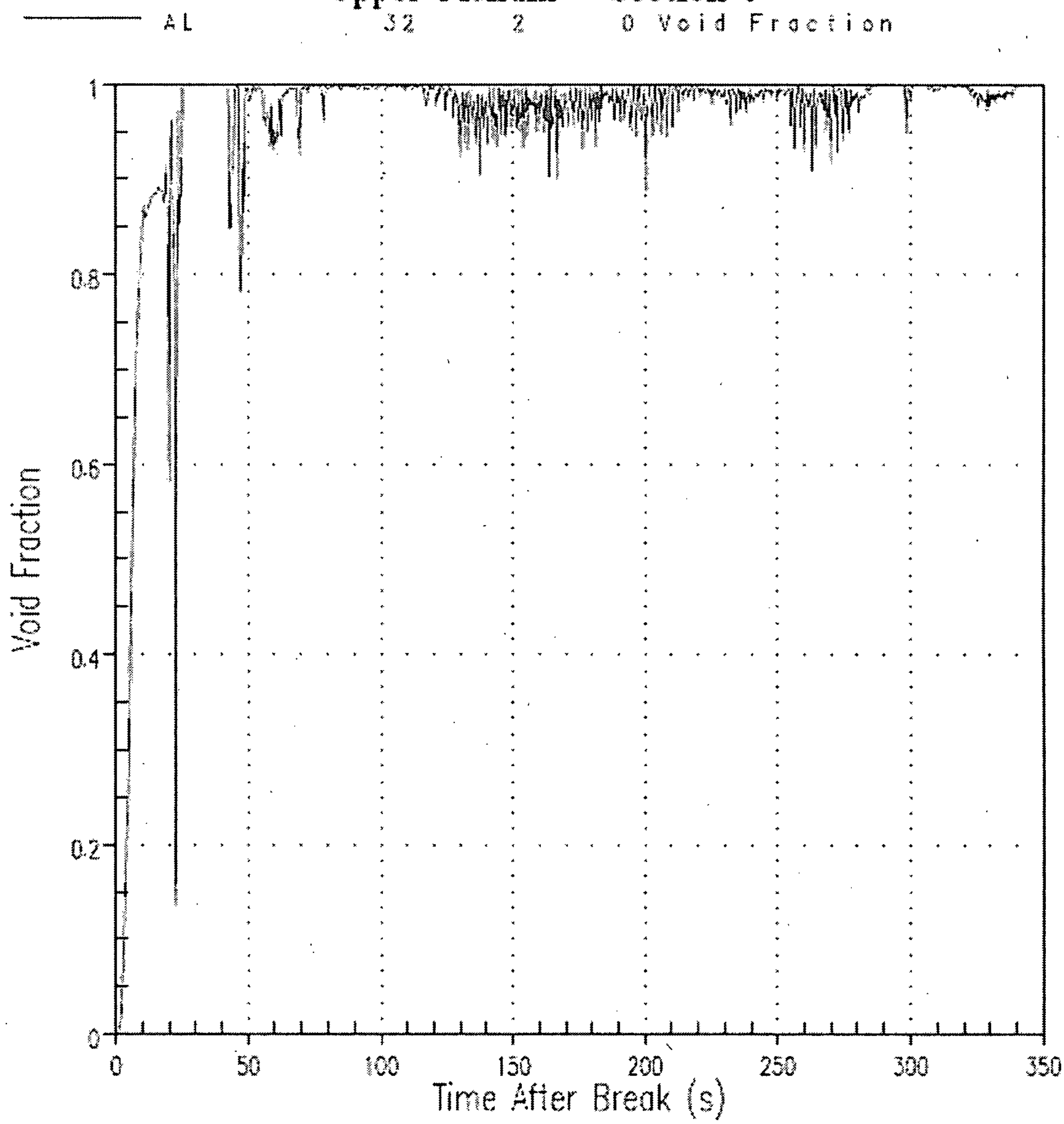
PTN Limiting PCT Case

Figure b69: Void Fraction from Intact Loop Downcomer Channel 78
Upper Plenum – Section 6



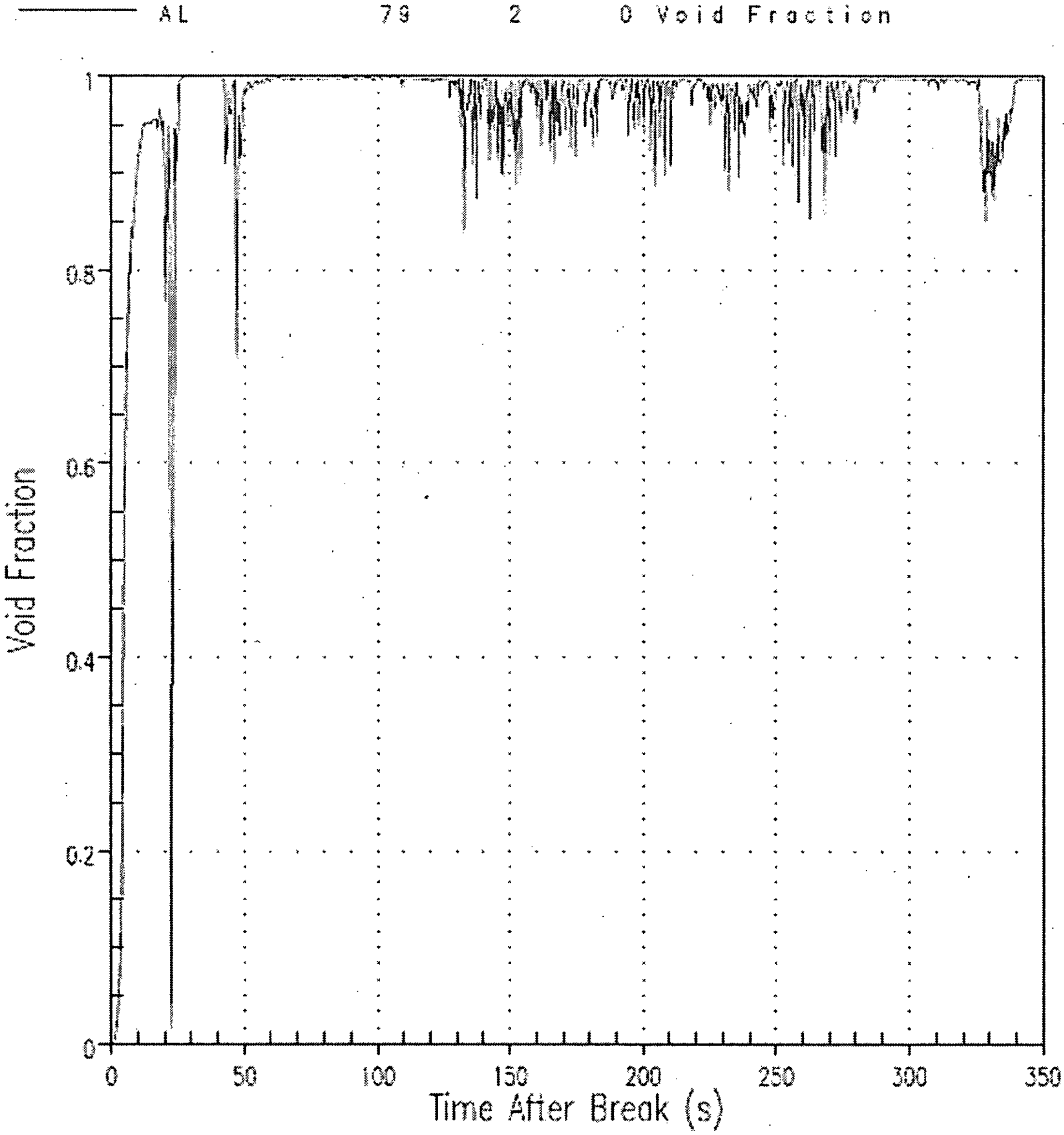
PTN Limiting PCT Case

Figure b70: Void Fraction from Broken Loop Downcomer Channel 32 Upper Plenum - Section 6



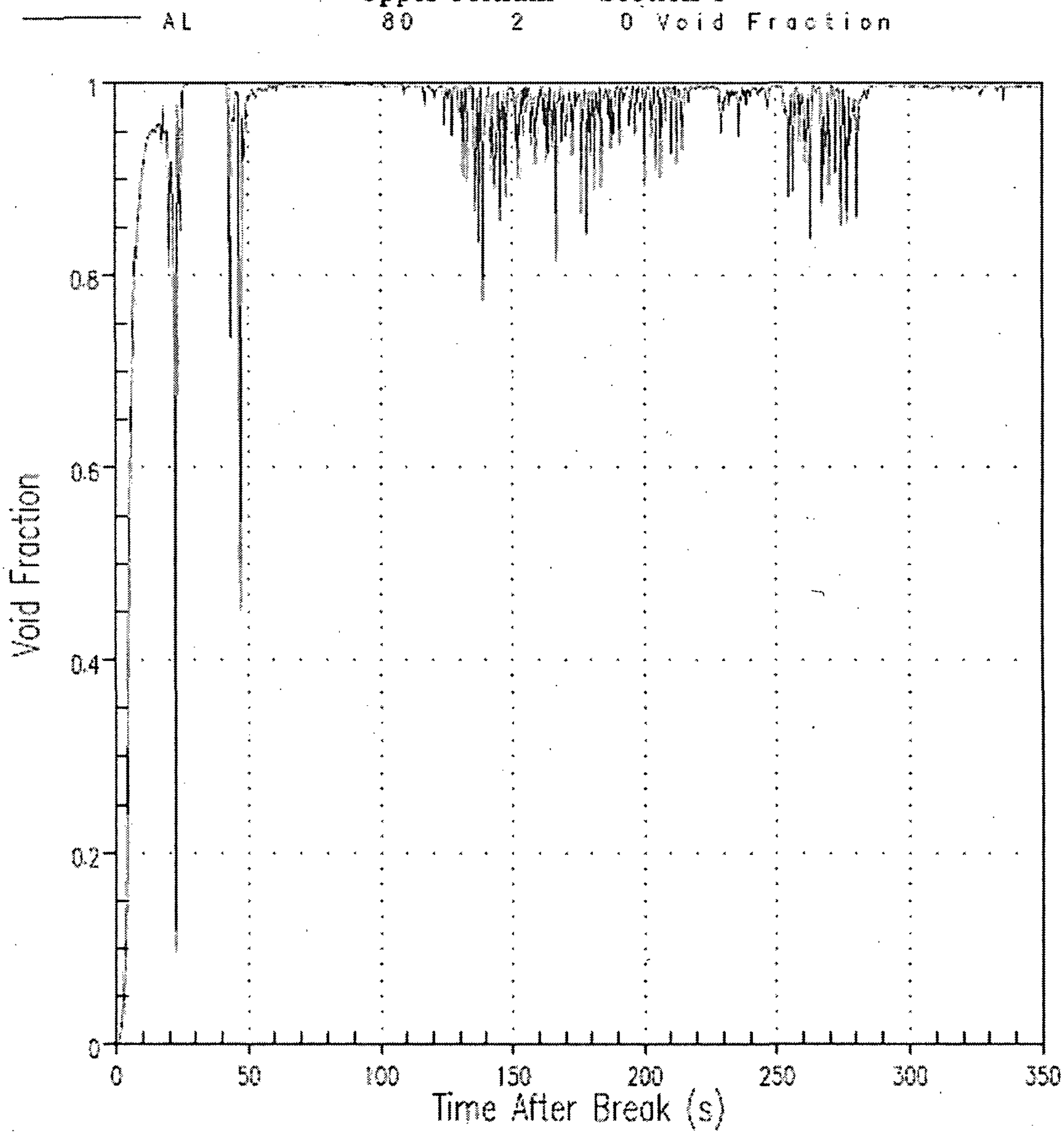
PTN Limiting PCT Case

Figure b71: Void Fraction from Adjacen to Broken Loop Downcomer Channel 79
Upper Plenum – Section 6



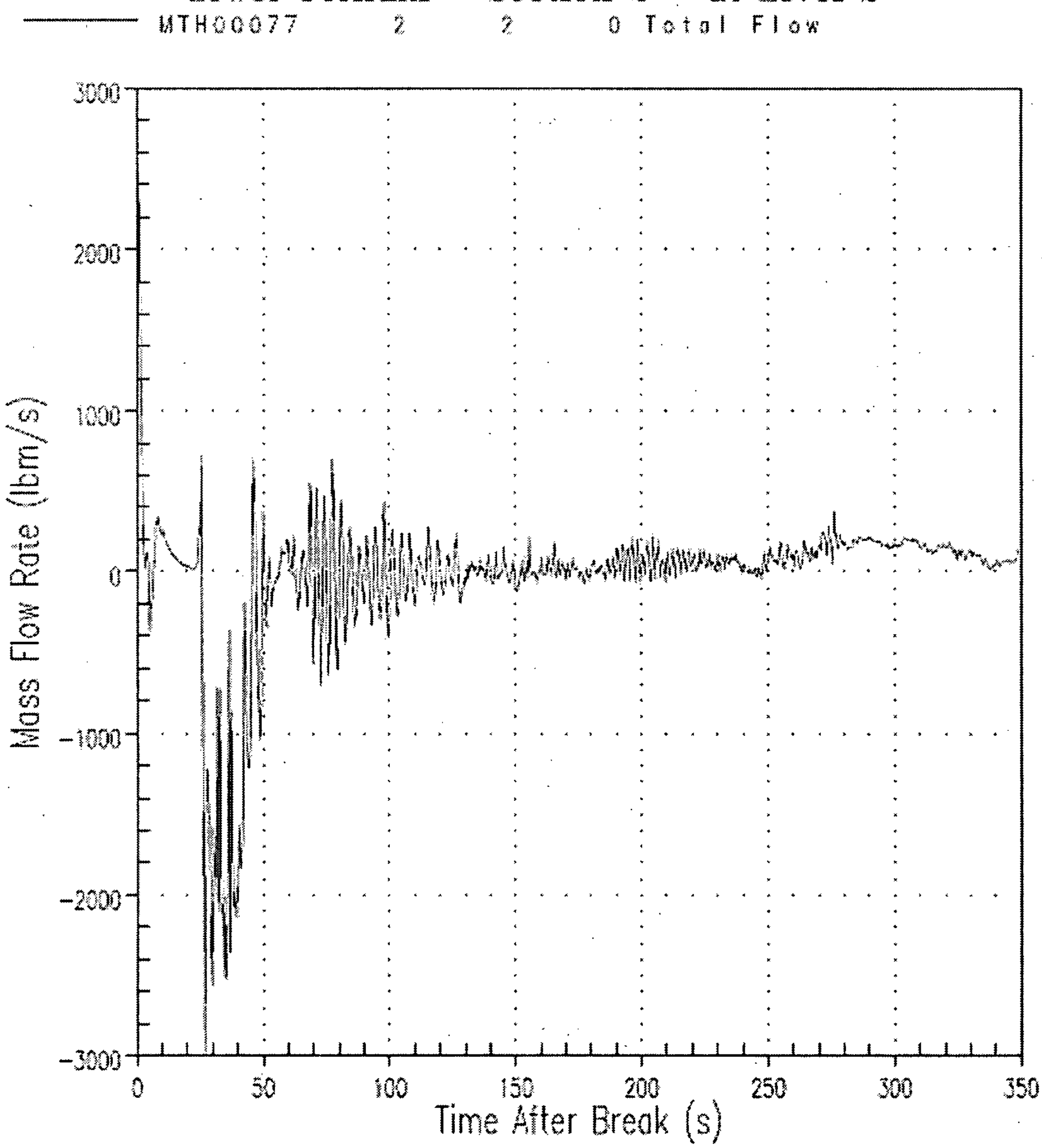
PTN Limiting PCT Case

Figure b72: Void Fraction from Adjacen to Broken Loop Downcomer Channel 80
Upper Plenum - Section 6



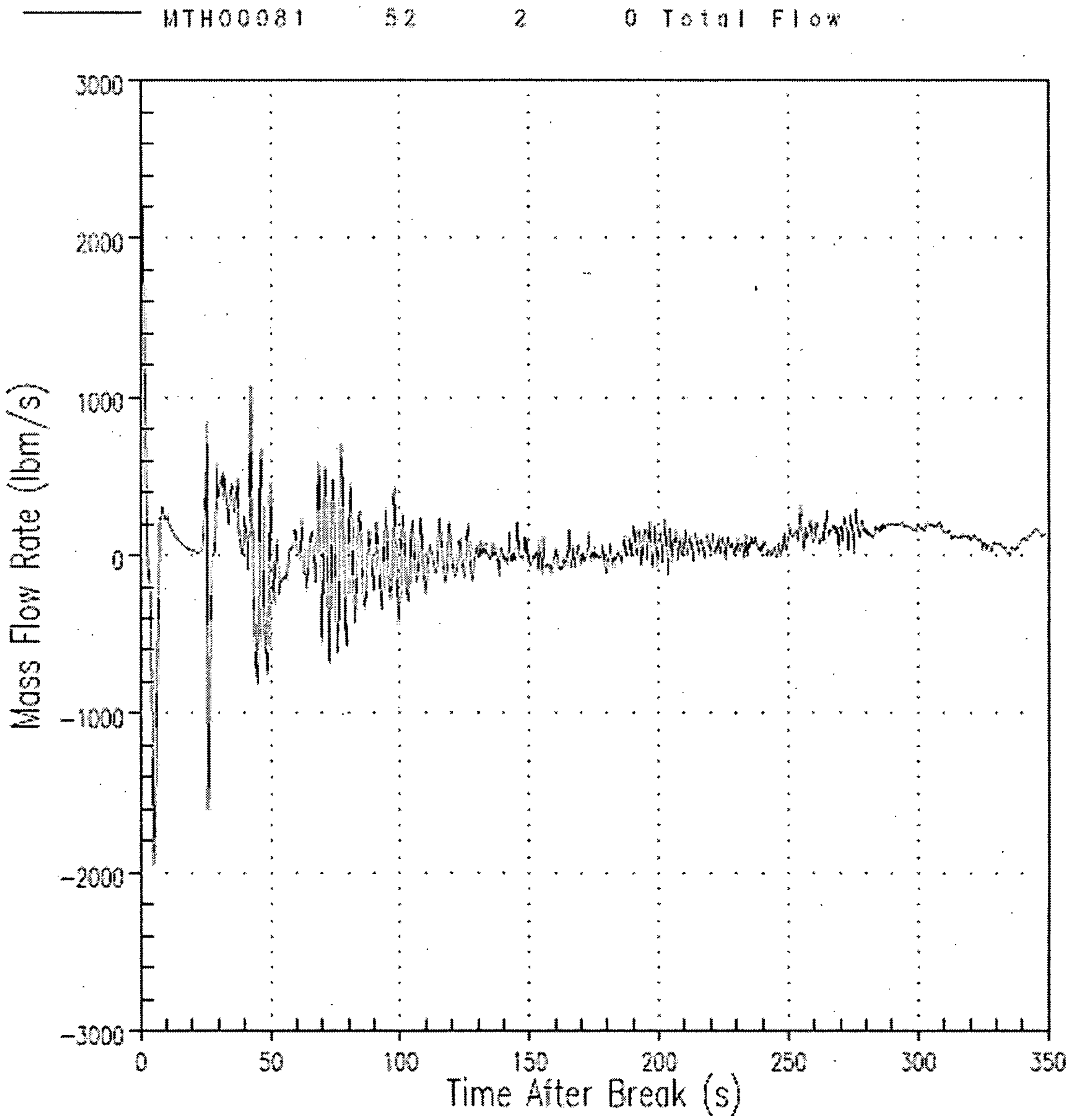
PTN Limiting PCT Case

Figure c1: Total Flow from Intact Loop Downcomer Channel 2
Lower Plenum – Section 2 – at Level 2



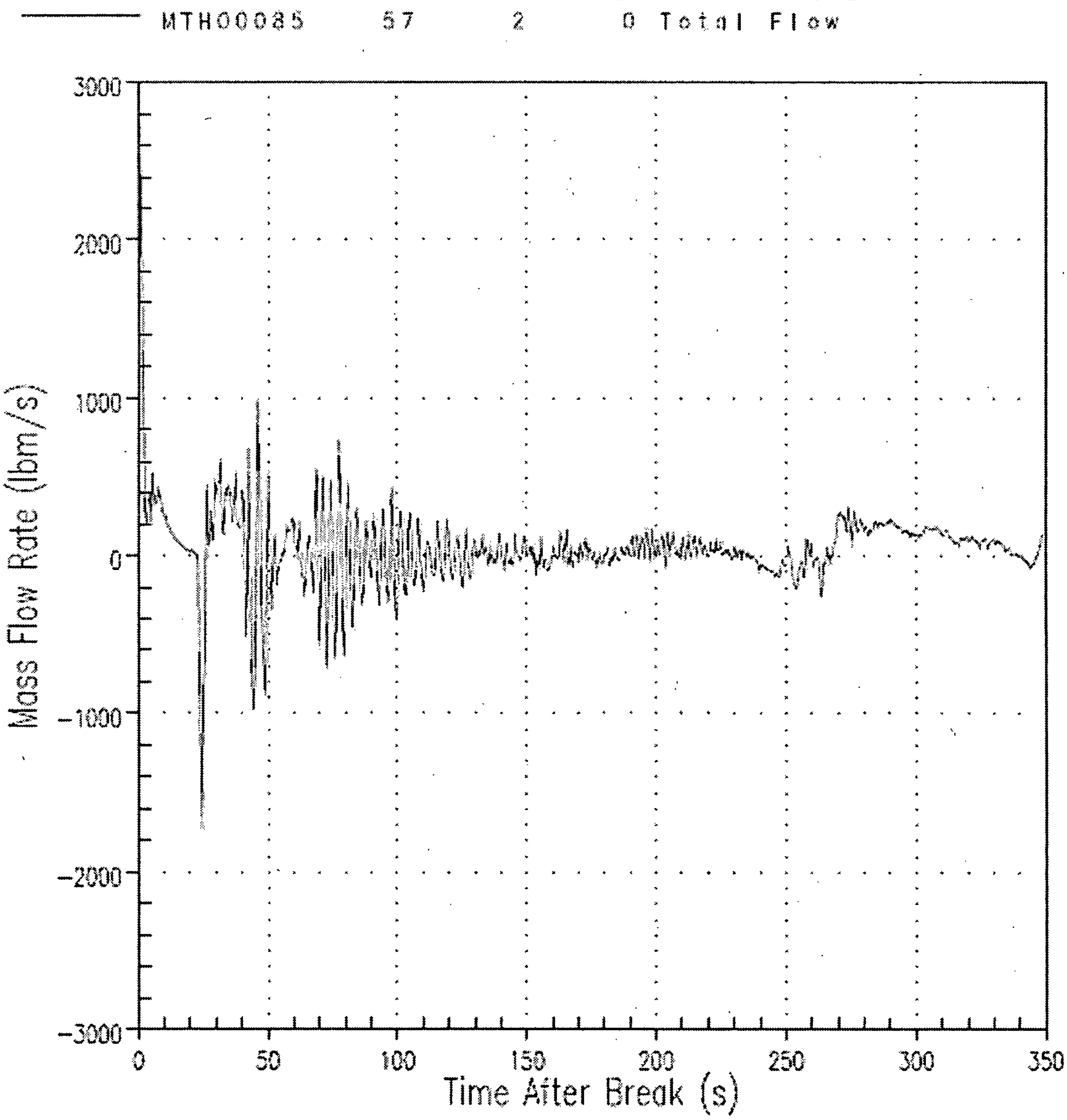
PTN Limiting PCT Case

Figure c2: Total Flow from Intact Loop Downcomer Channel 52
Lower Plenum - Section 2 - at Level 2



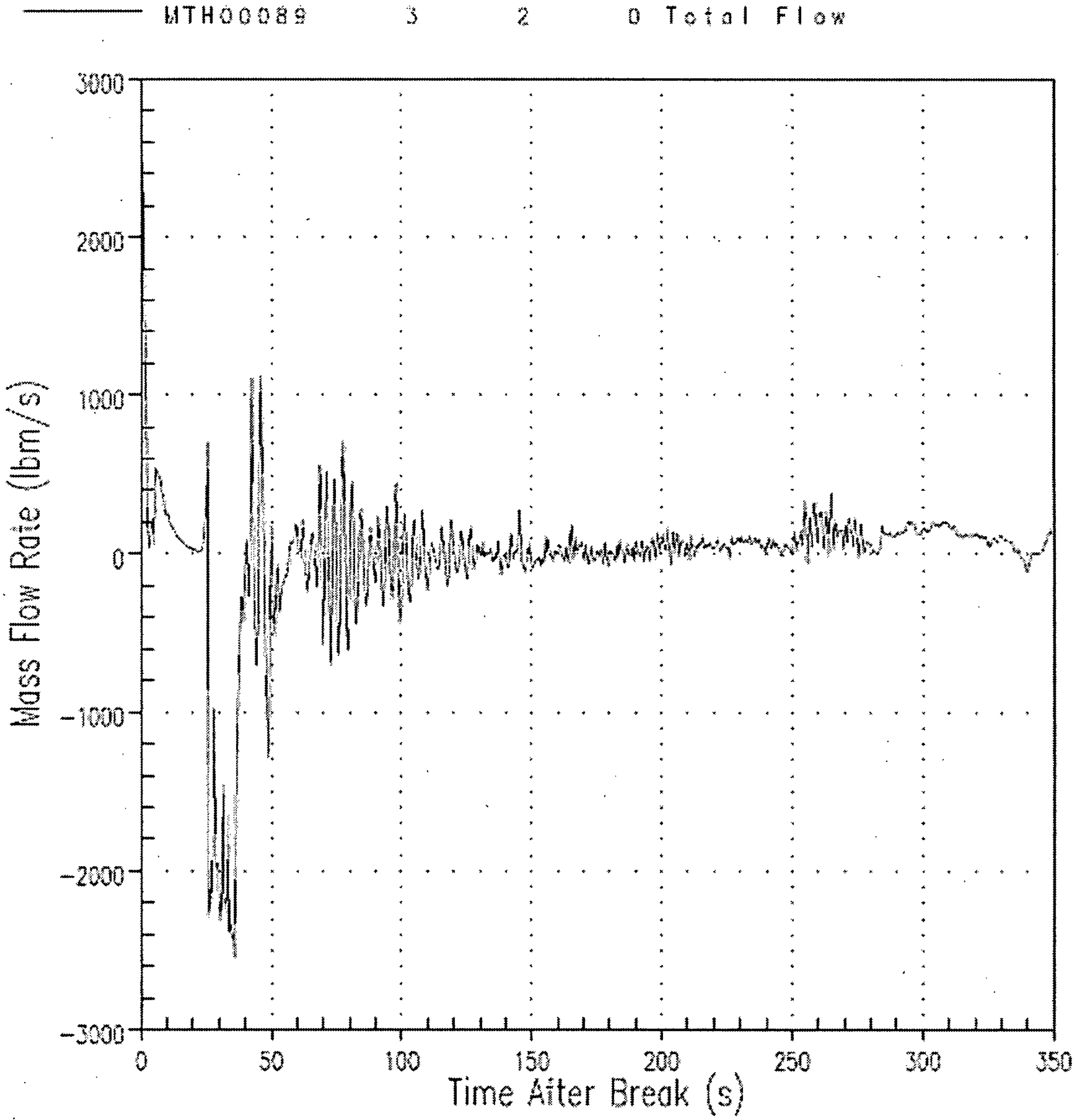
PTN Limiting PCT Case

Figure c3: Total Flow from Intact Loop Downcomer Channel 57 Lower Plenum – Section 2 – at Level 2



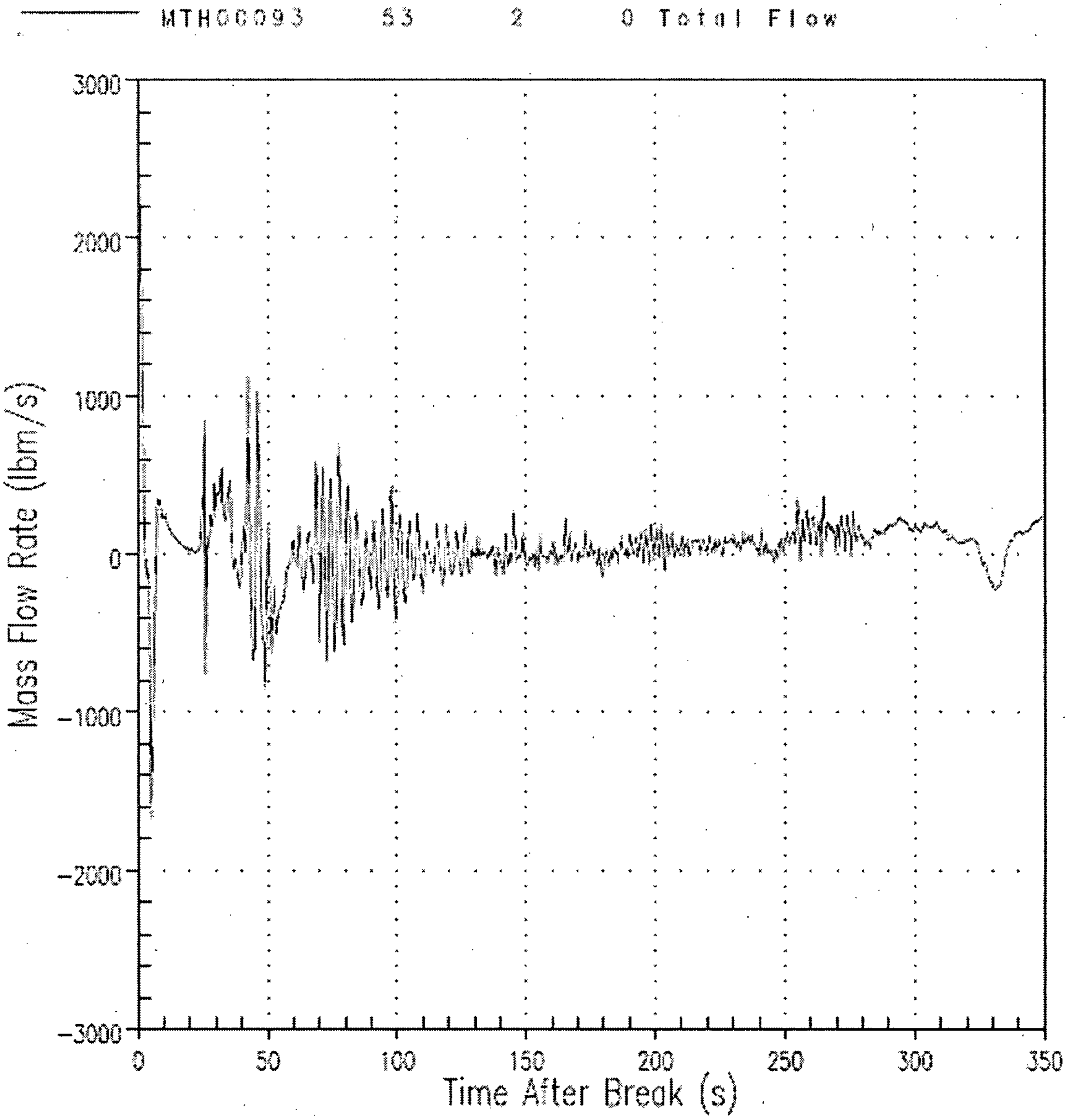
PTN Limiting PCT Case

Figure c4: Total Flow from Intact Loop Downcomer Channel 3
Lower Plenum - Section 2 - at Level 2



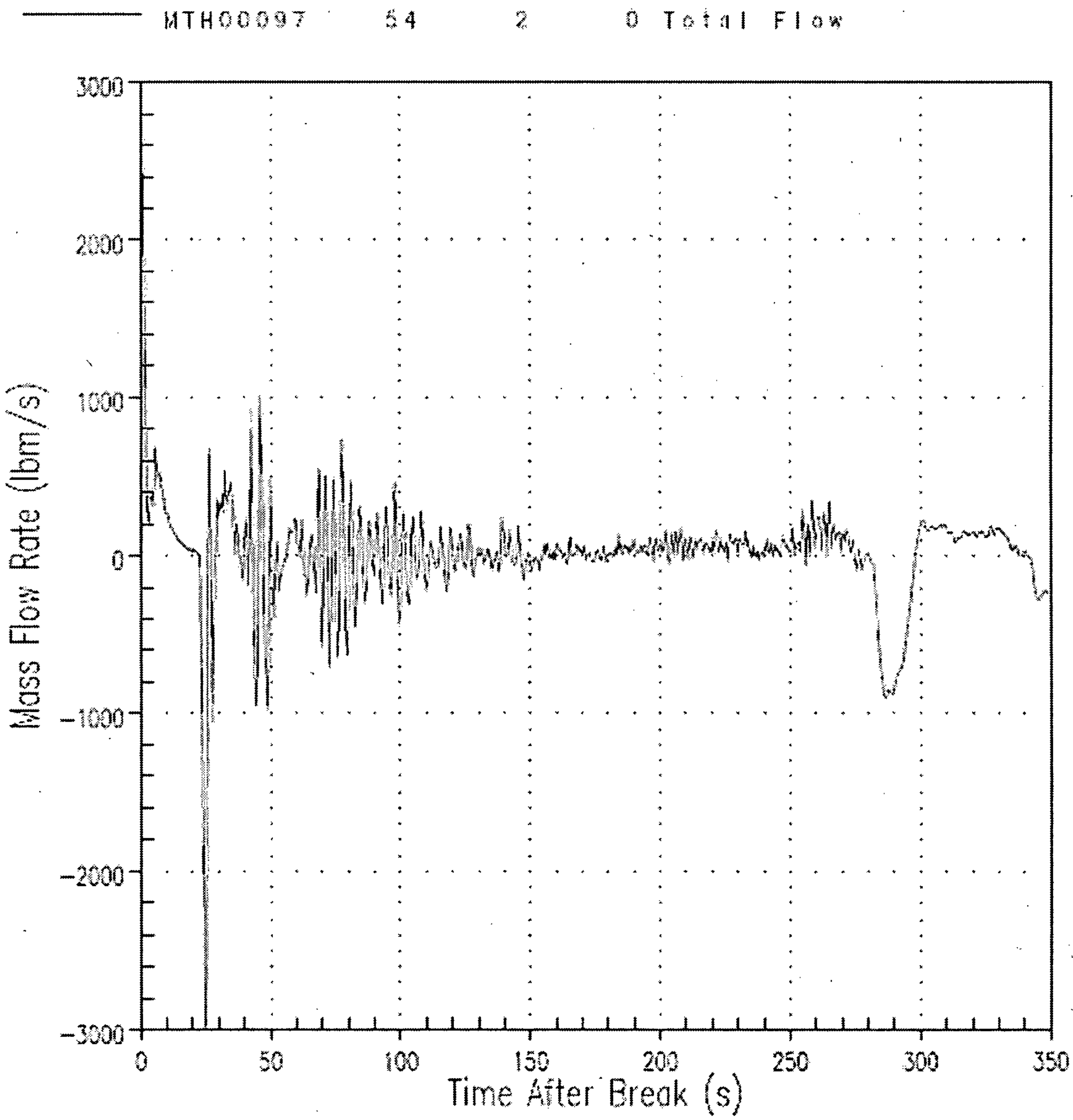
PTN Limiting PCT Case

Figure c5: Total Flow from Intact Loop Downcomer Channel 53 Lower Plenum – Section 2 – at Level 2



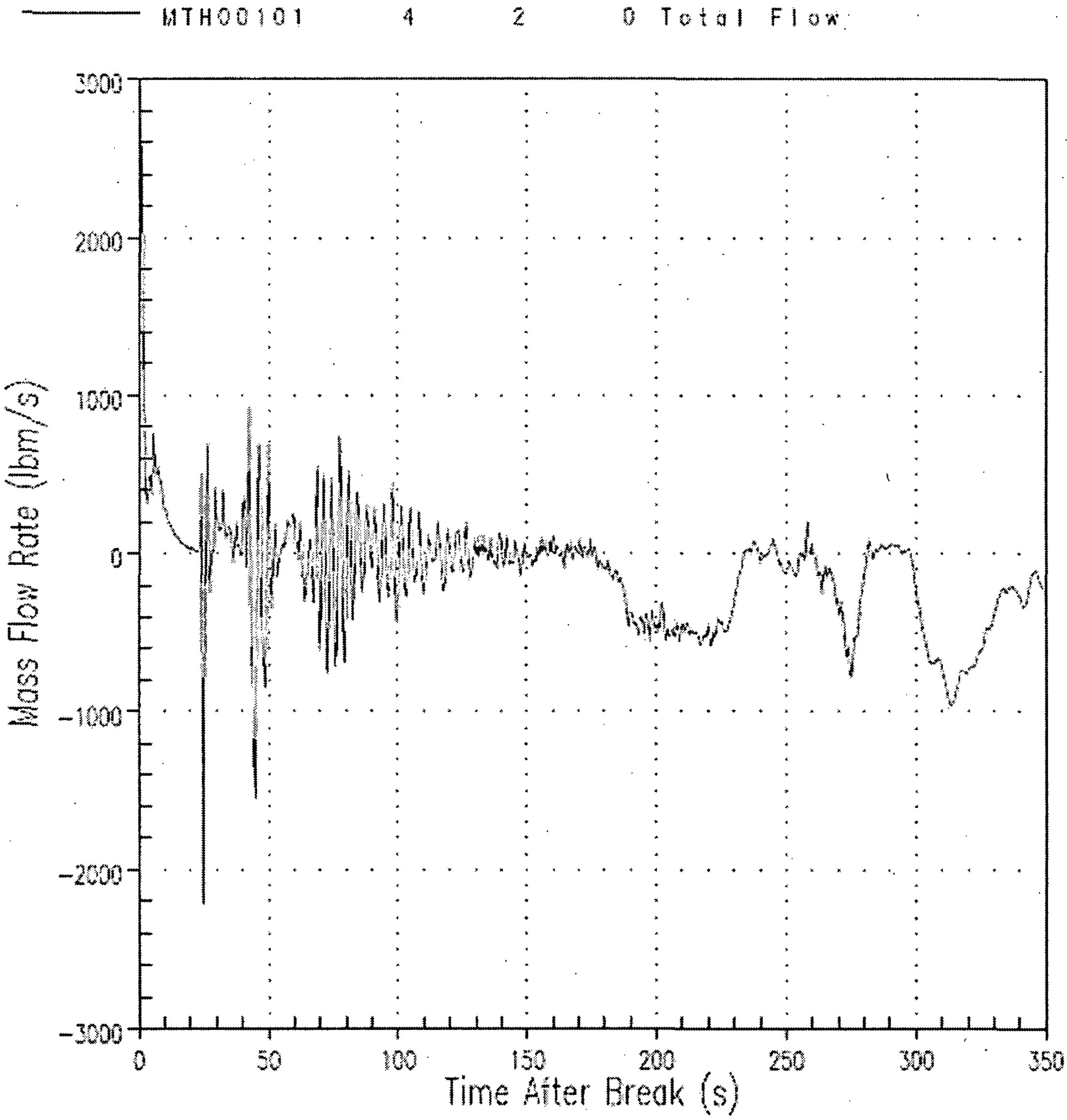
PTN Limiting PCT Case

Figure c6: Total Flow from Intact Loop Downcomer Channel 54
Lower Plenum – Section 2 – at Level 2



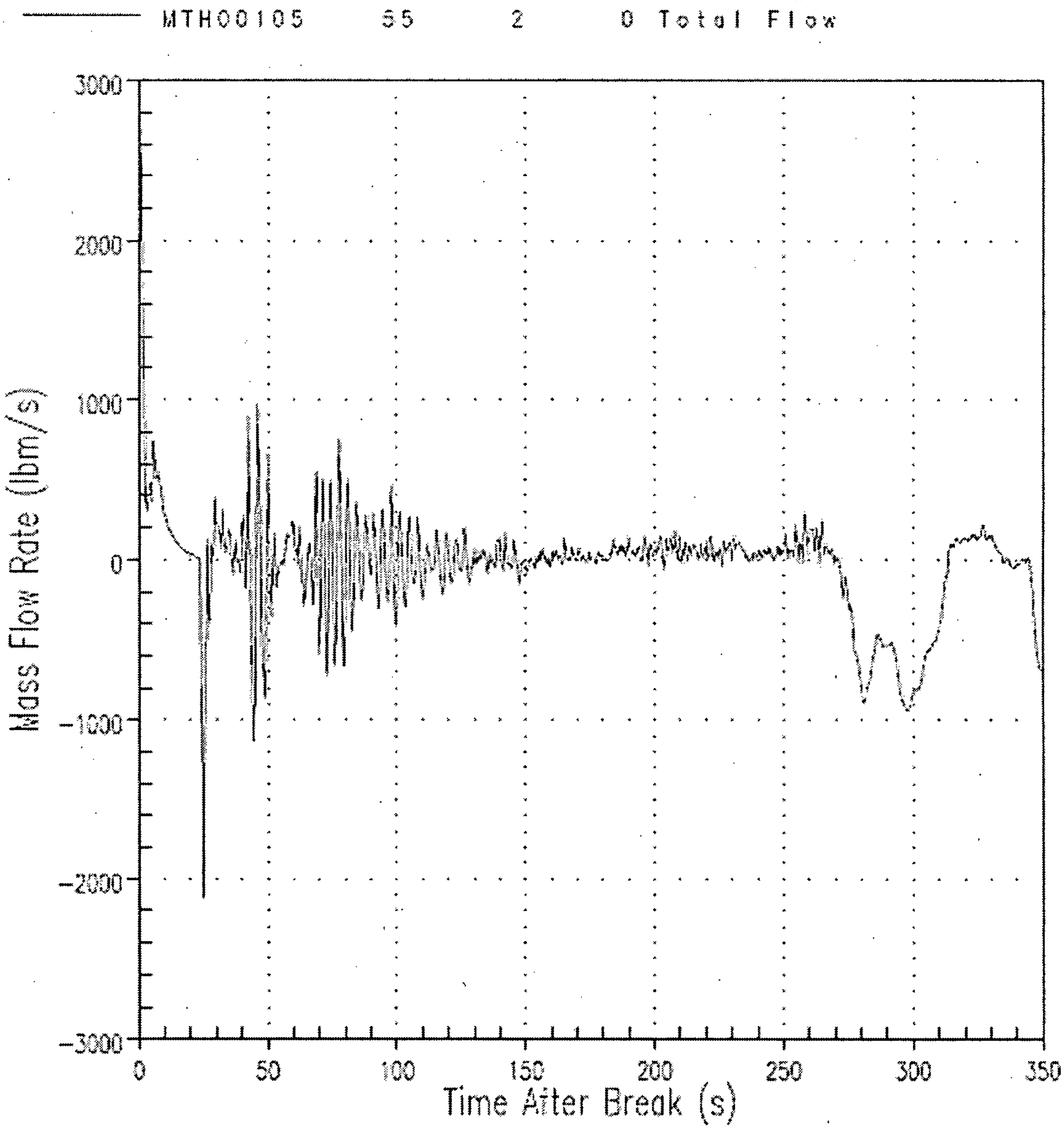
PTN Limiting PCT Case

Figure c7: Total Flow from Broken Loop Downcomer Channel 4
Lower Plenum – Section 2 – at Level 2



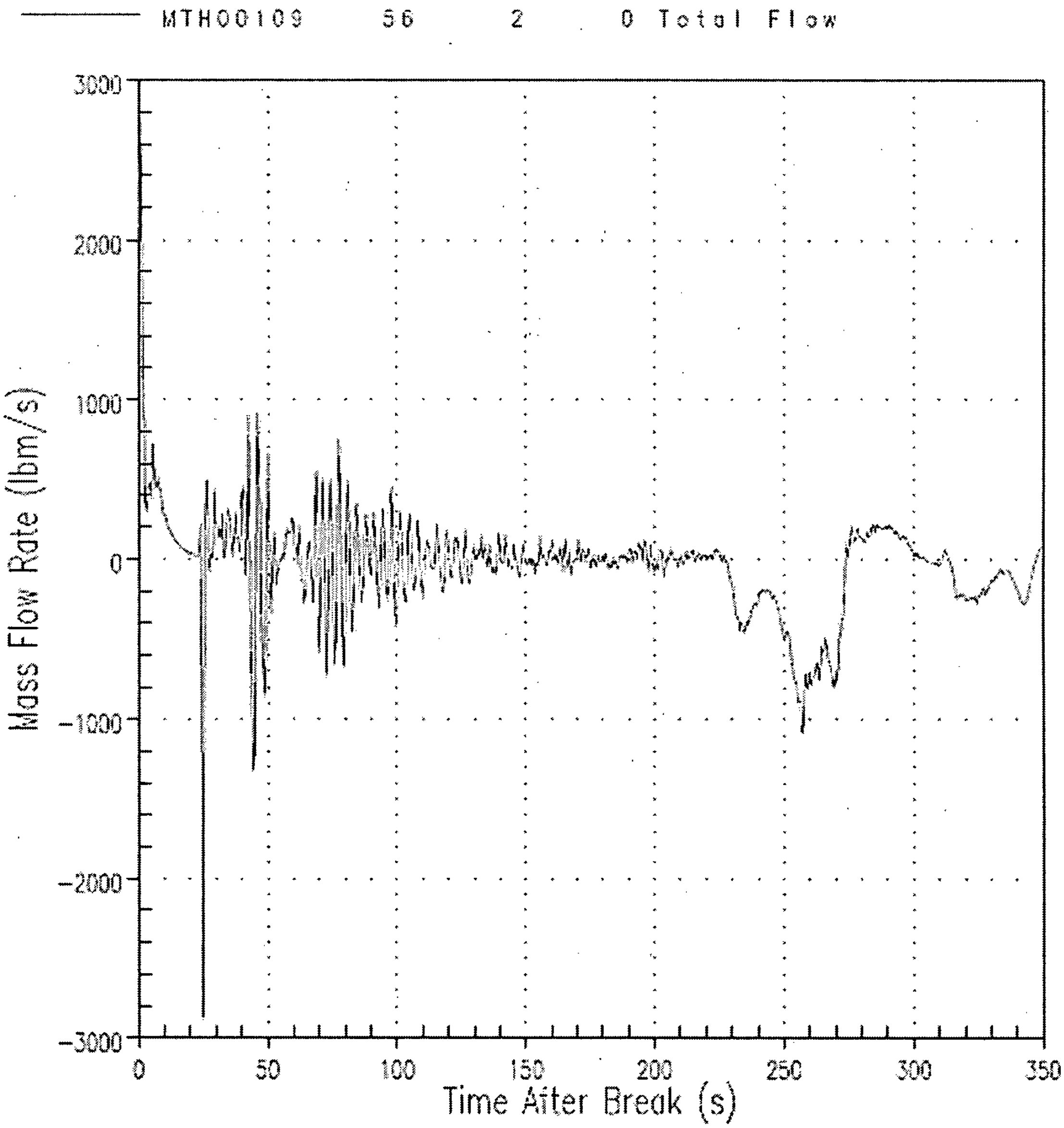
PTN Limiting PCT Case

Figure c8: Total Flow from Adjacen to Broken Loop Downcomer Channel 55 Lower Plenum - Section 2 - at Level 2



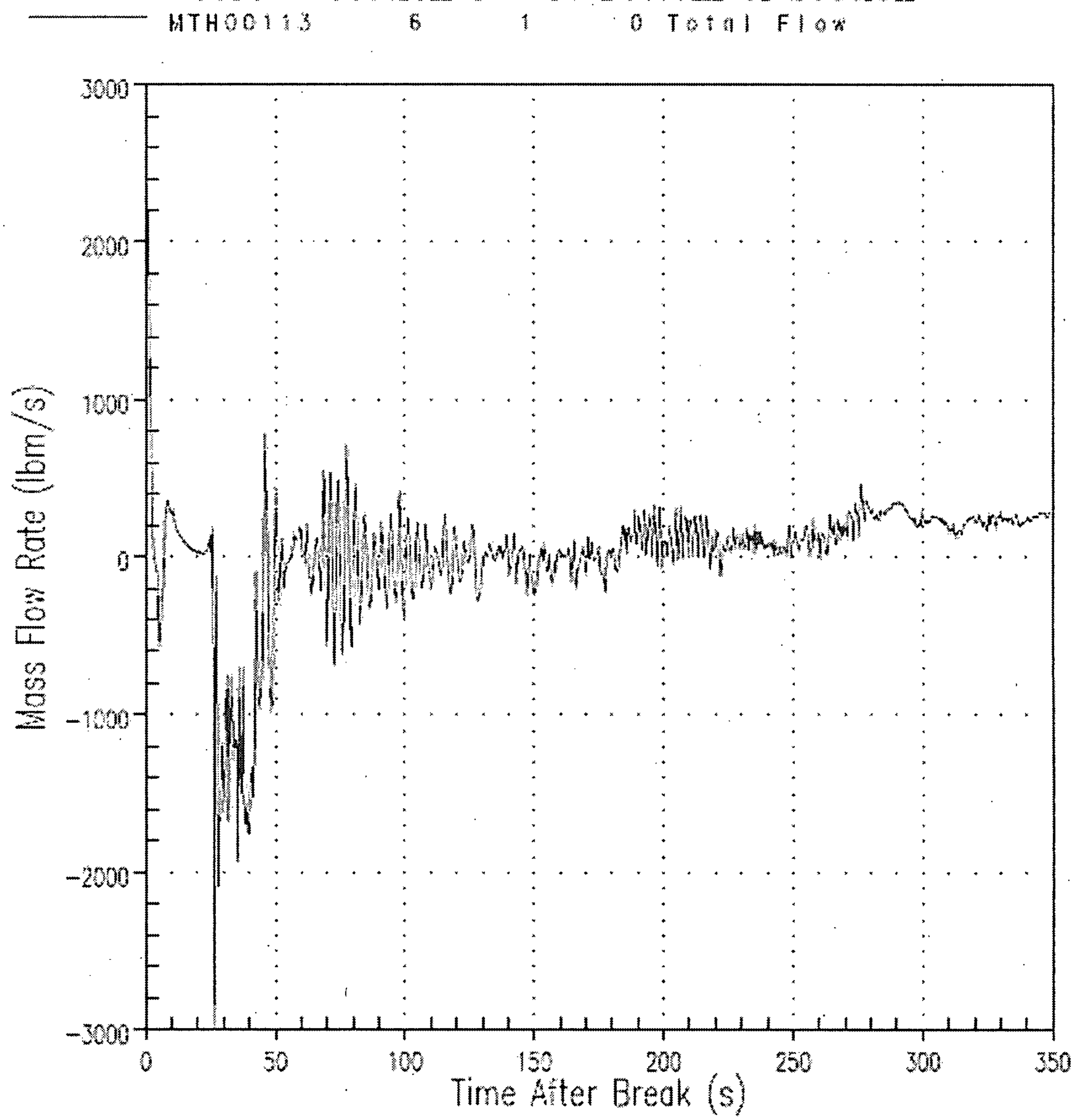
PTN Limiting PCT Case

Figure c9: Total Flow from Adjacen to Broken Loop Downcomer Channel 56 Lower Plenum - Section 2 - at Level 2



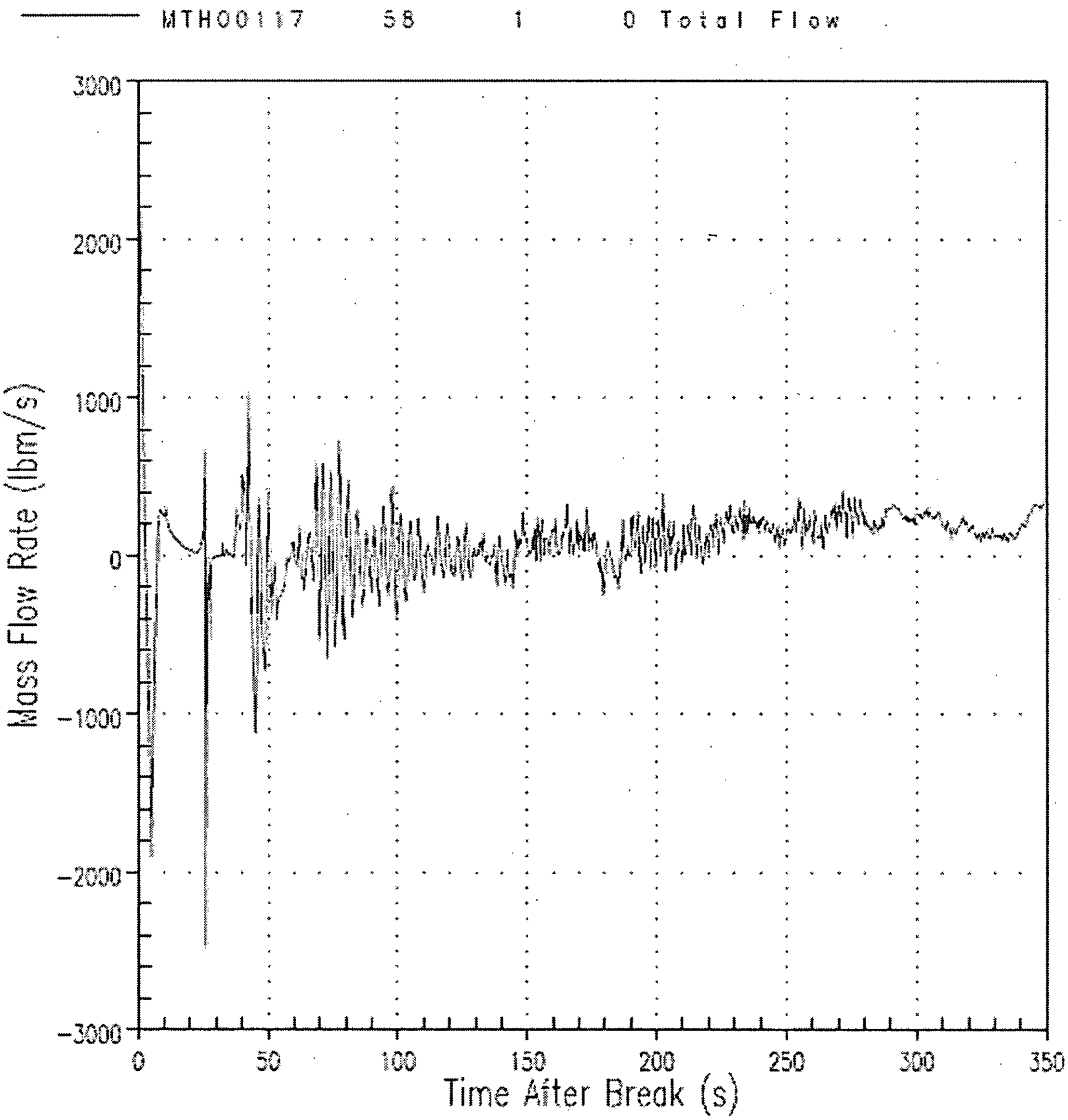
PTN Limiting PCT Case

Figure c10: Total Flow from Intact Loop Downcomer Channel 6
Core – Section 3 – at Bottom of Section



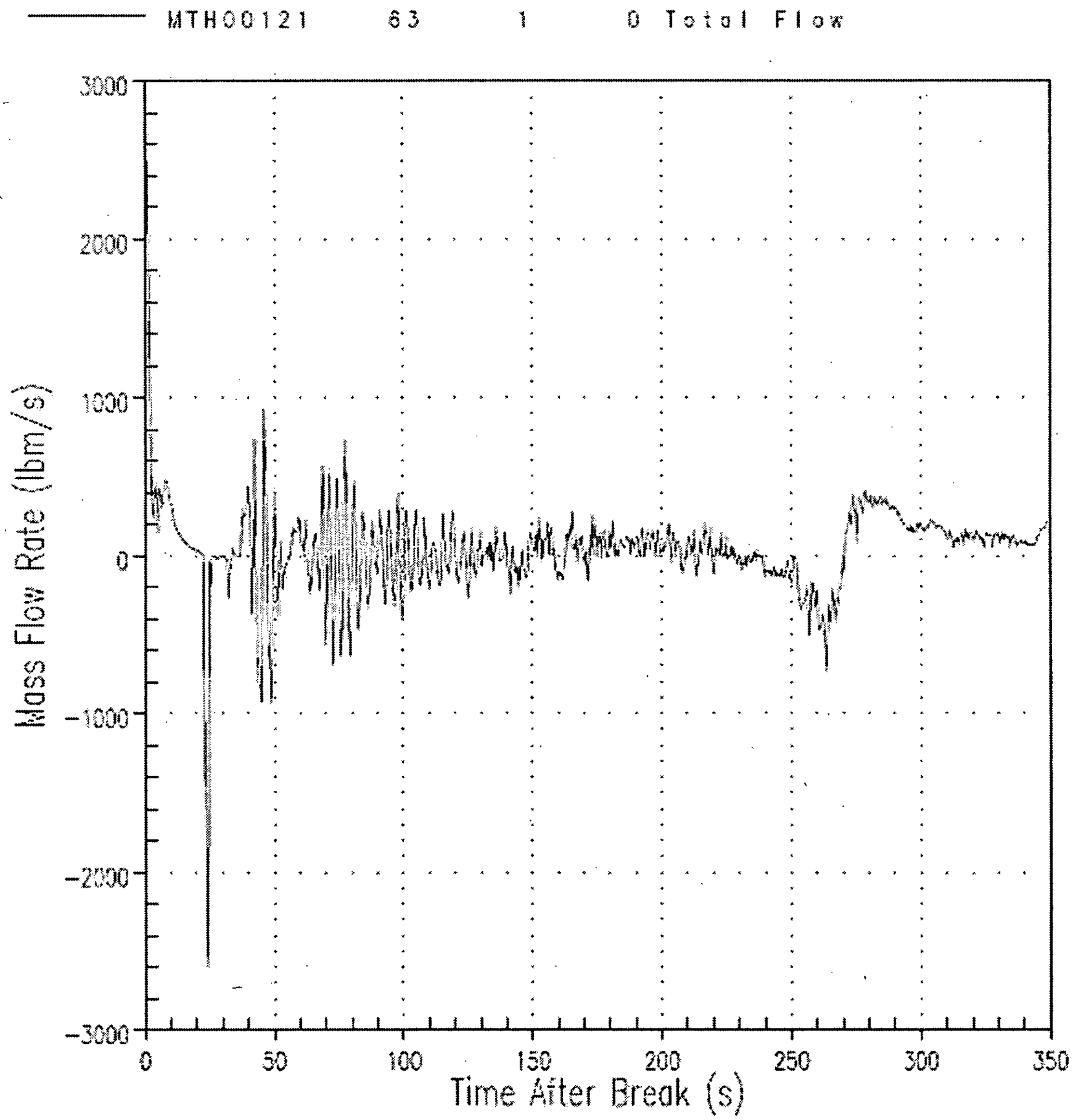
PTN Limiting PCT Case

Figure c11: Total Flow from Intact Loop Downcomer Channel 58 Core – Section 3 – at Bottom of Section



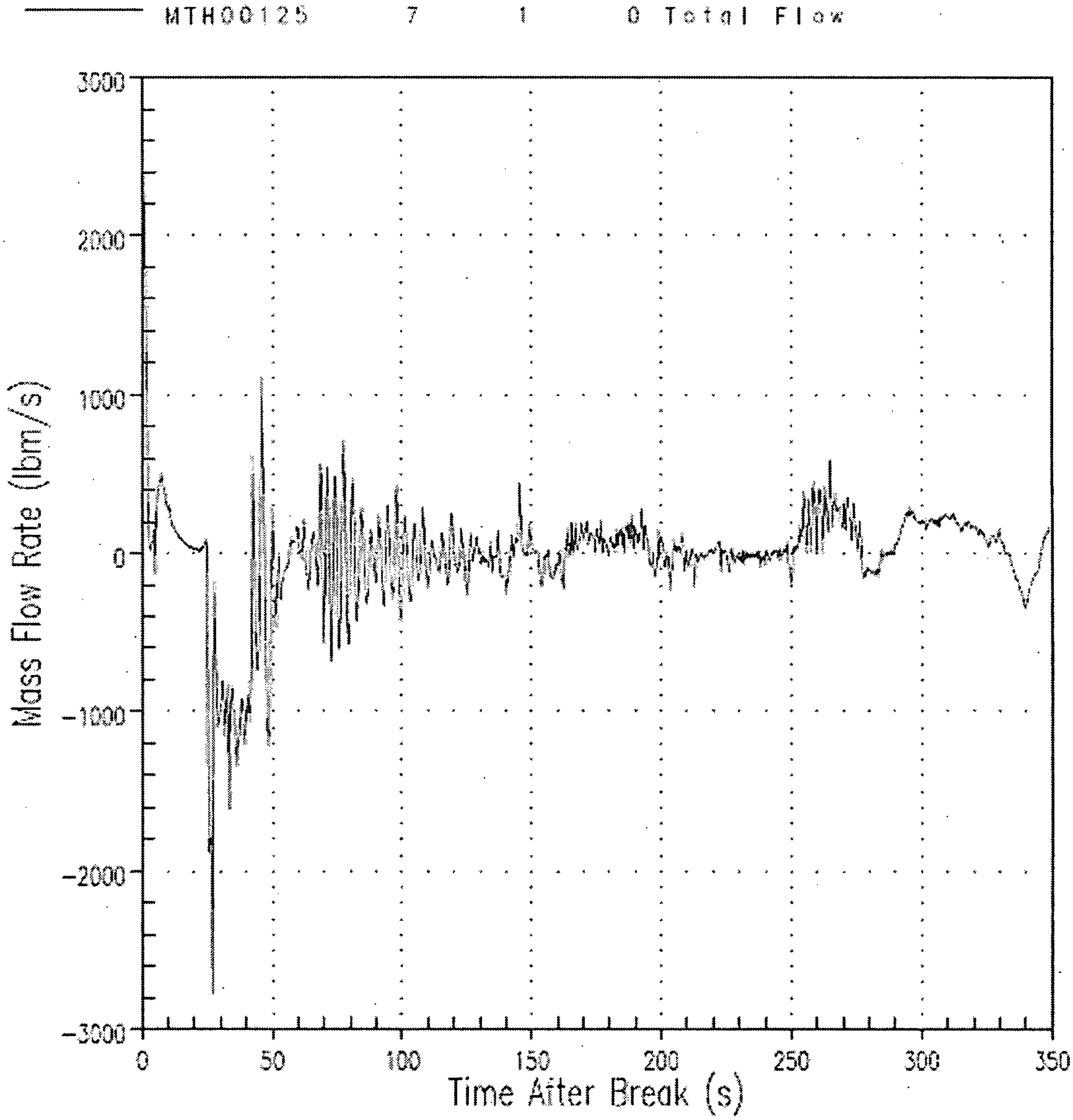
PTN Limiting PCT Case

Figure c12: Total Flow from Intact Loop Downcomer Channel 63 Core – Section 3 – at Bottom of Section



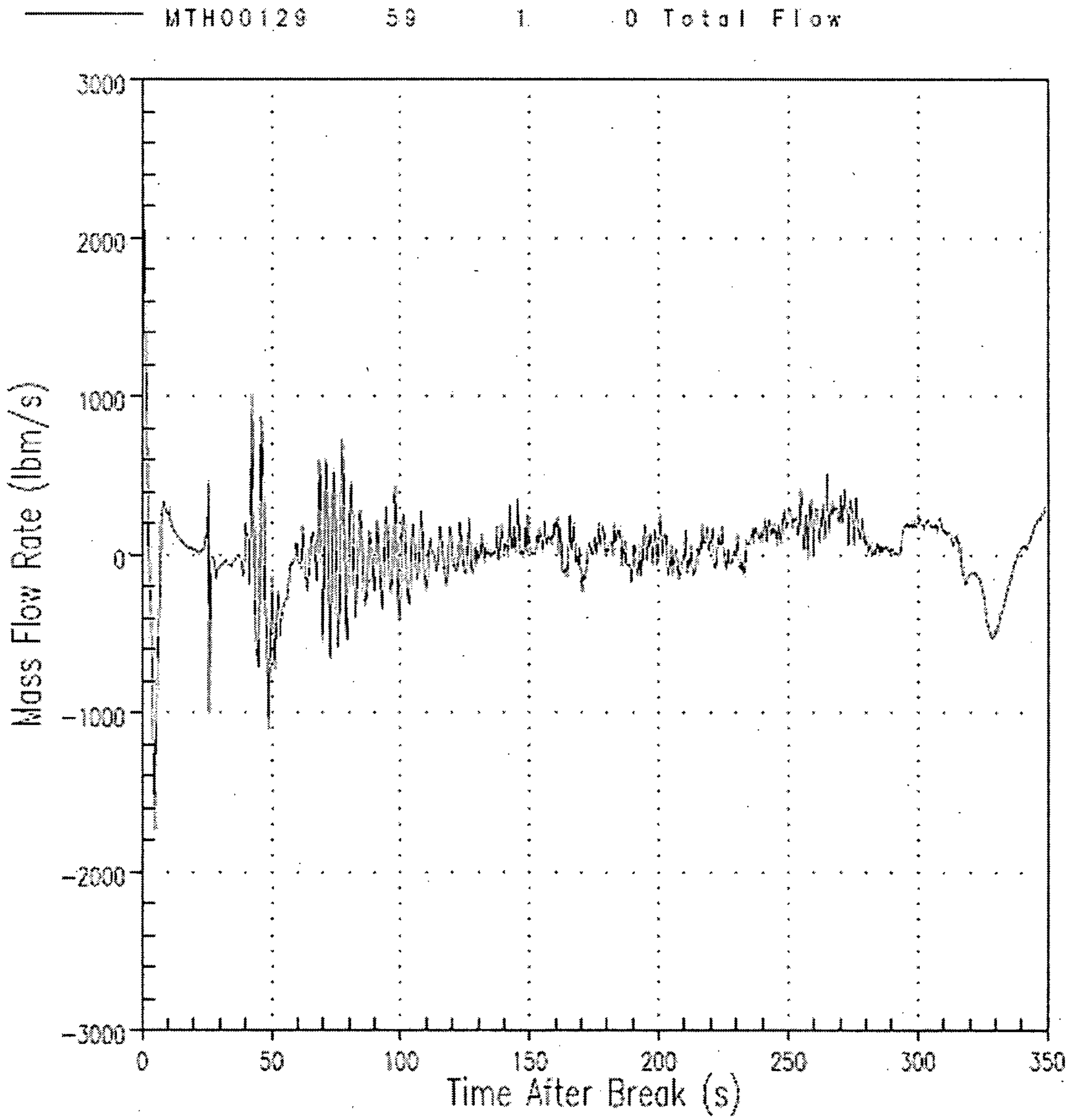
PTN Limiting PCT Case

Figure c13: Total Flow from Intact Loop Downcomer Channel 7 Core – Section 3 – at Bottom of Section

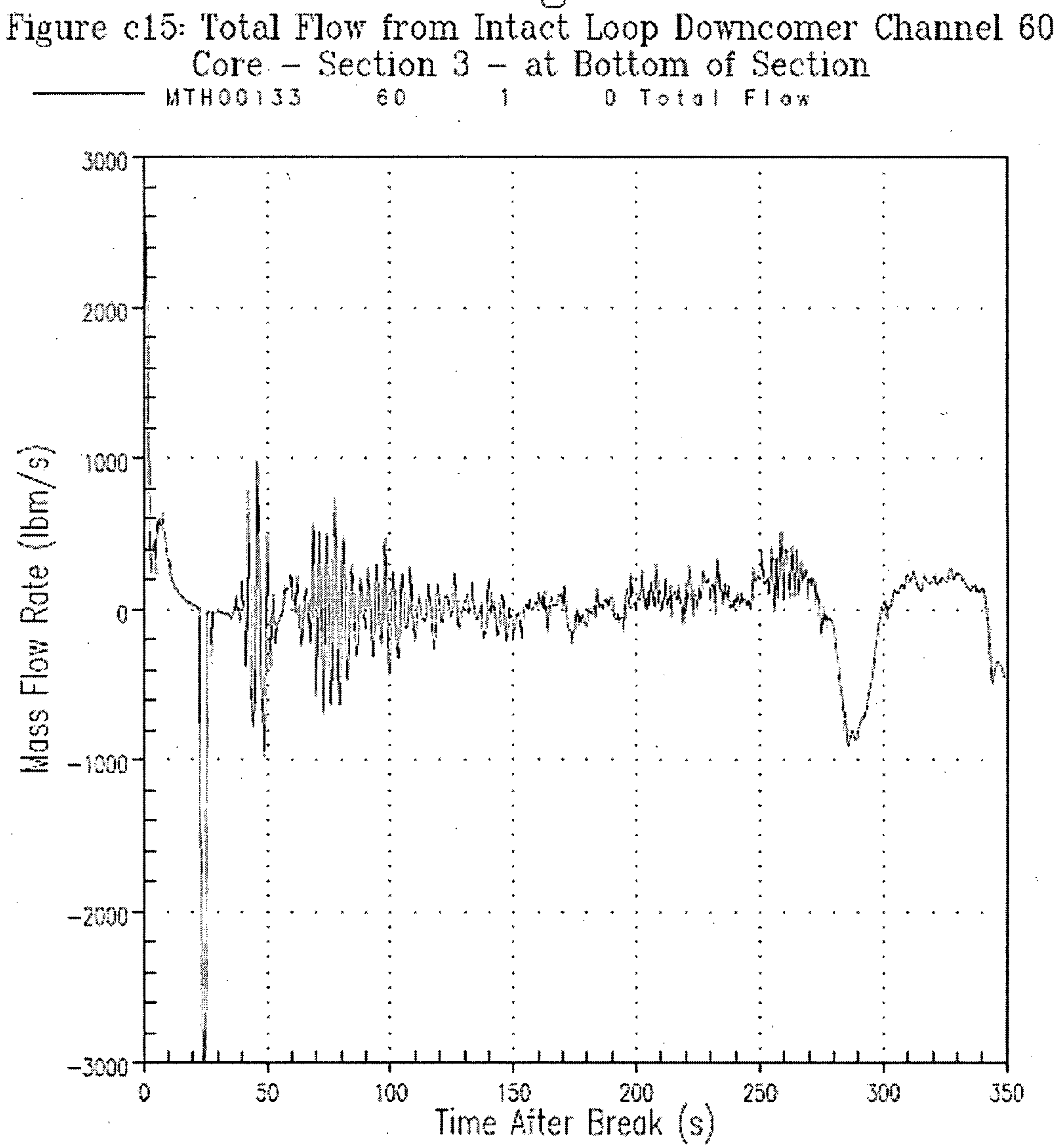


PTN Limiting PCT Case

Figure c14: Total Flow from Intact Loop Downcomer Channel 59
Core – Section 3 – at Bottom of Section

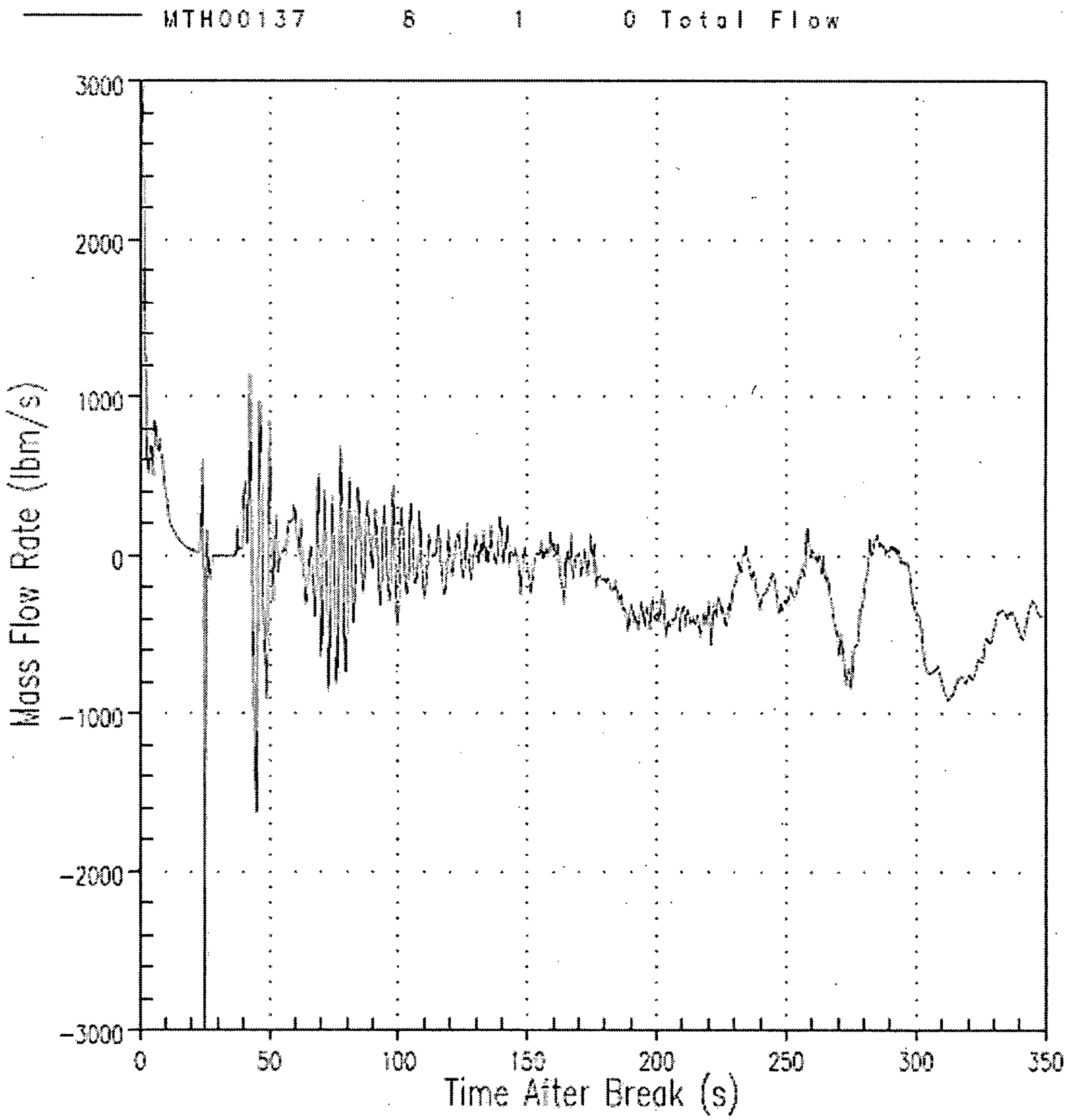


PTN Limiting PCT Case



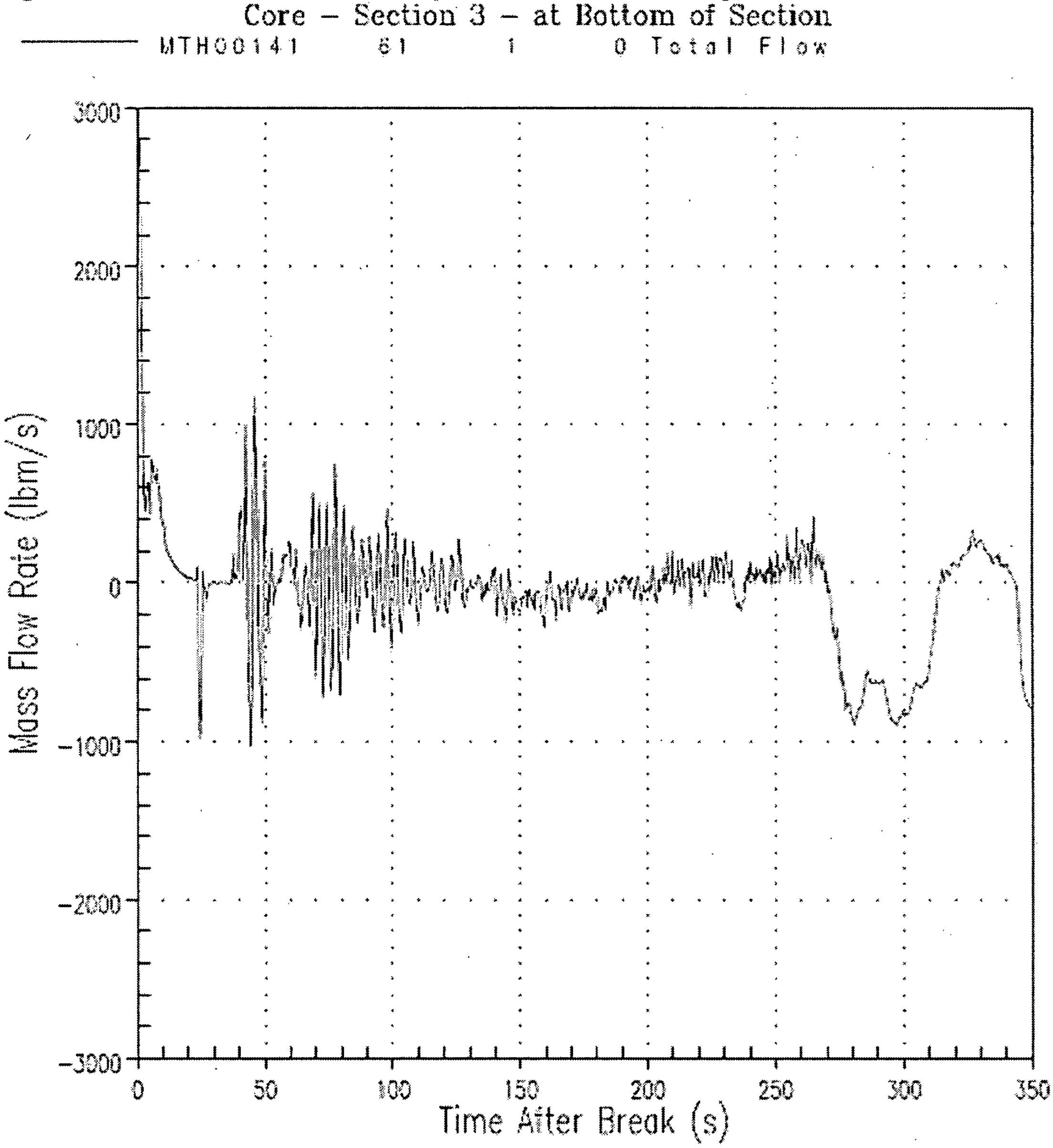
PTN Limiting PCT Case

Figure c16: Total Flow from Broken Loop Downcomer Channel 8
Core – Section 3 – at Bottom of Section



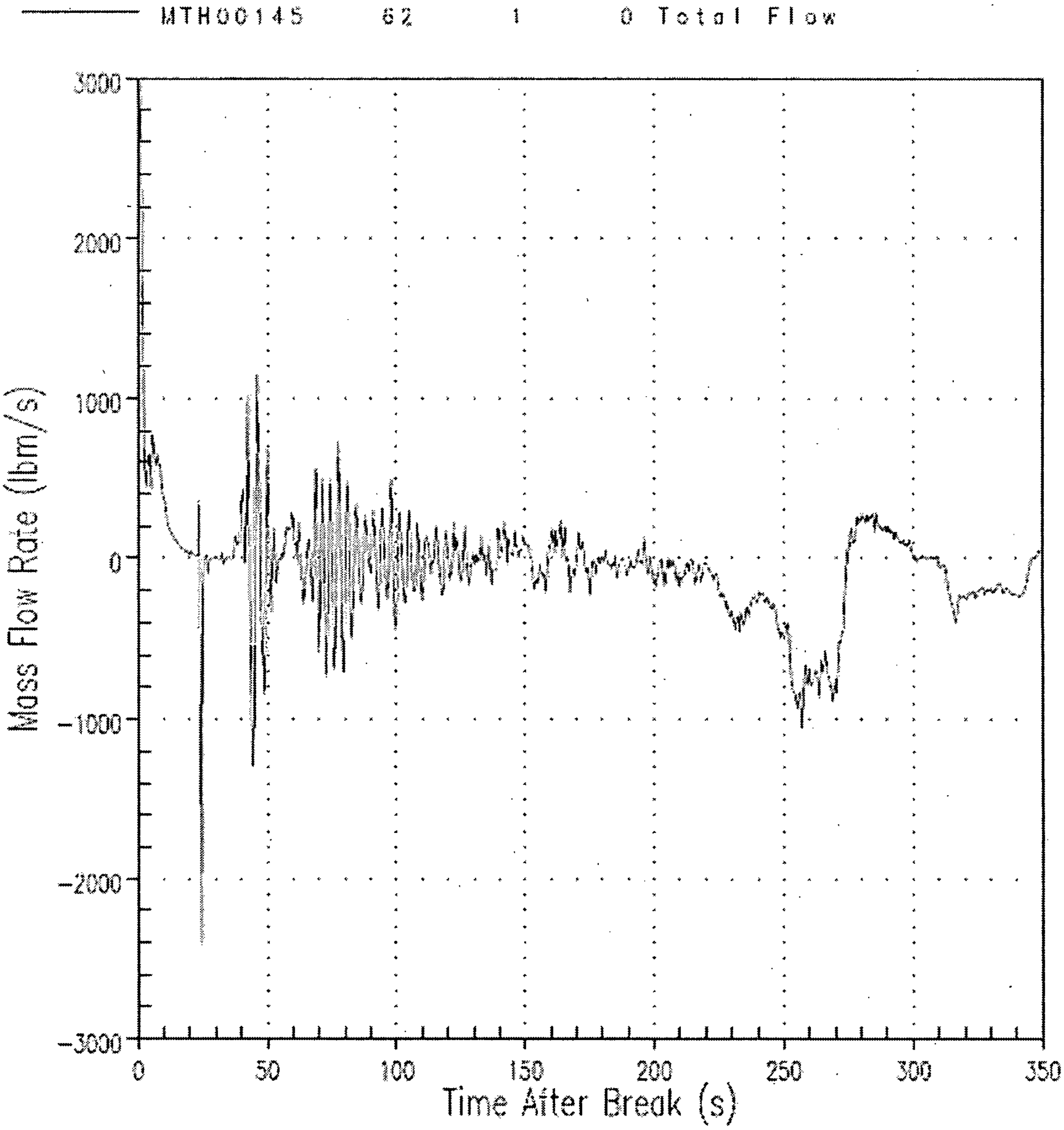
PTN Limiting PCT Case

Figure c17: Total Flow from Adjacen to Broken Loop Downcomer Channel 61 Core – Section 3 – at Bottom of Section



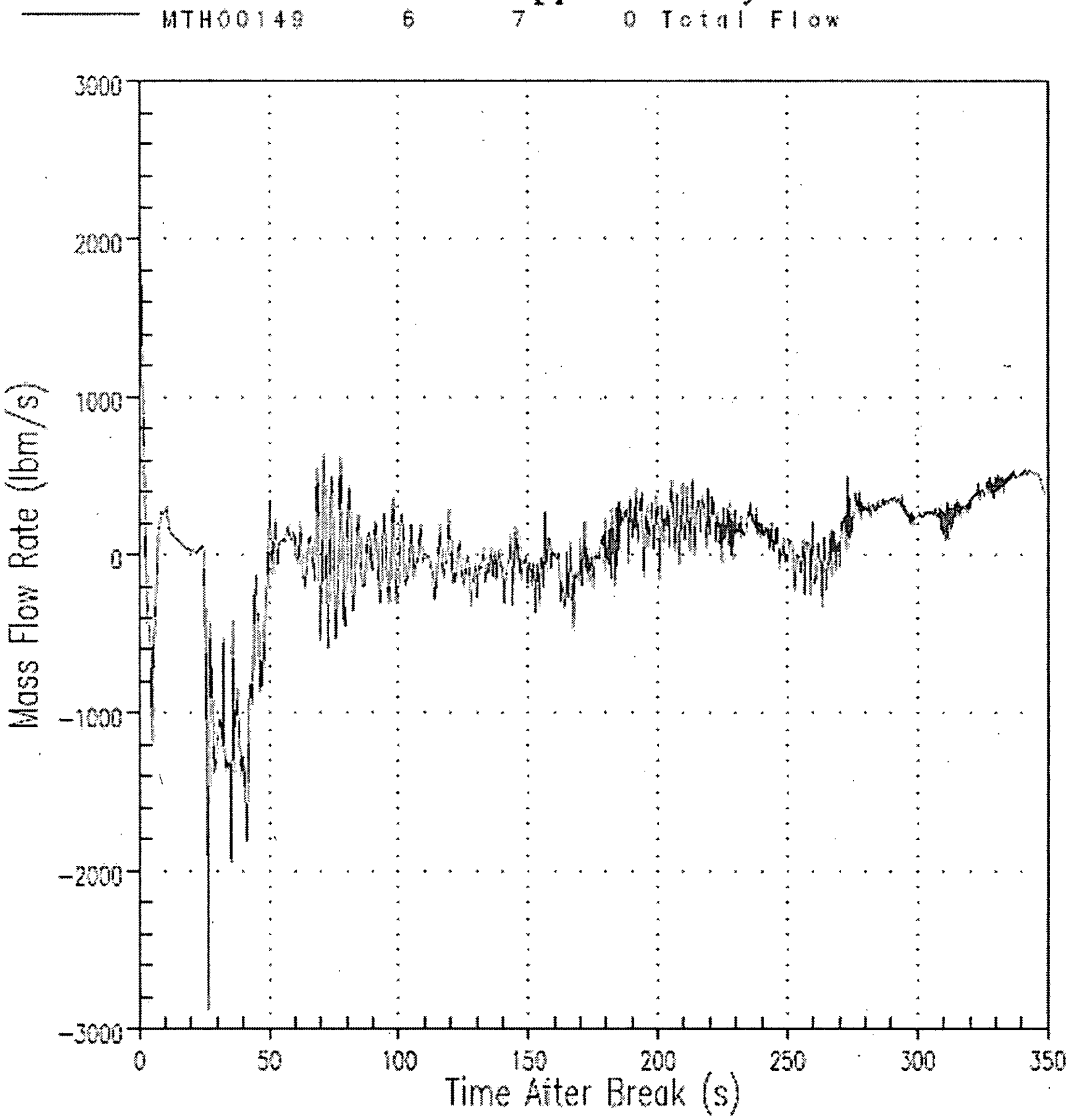
PTN Limiting PCT Case

Figure c18: Total Flow from Adjacen to Broken Loop Downcomer Channel 62
Core – Section 3 – at Bottom of Section



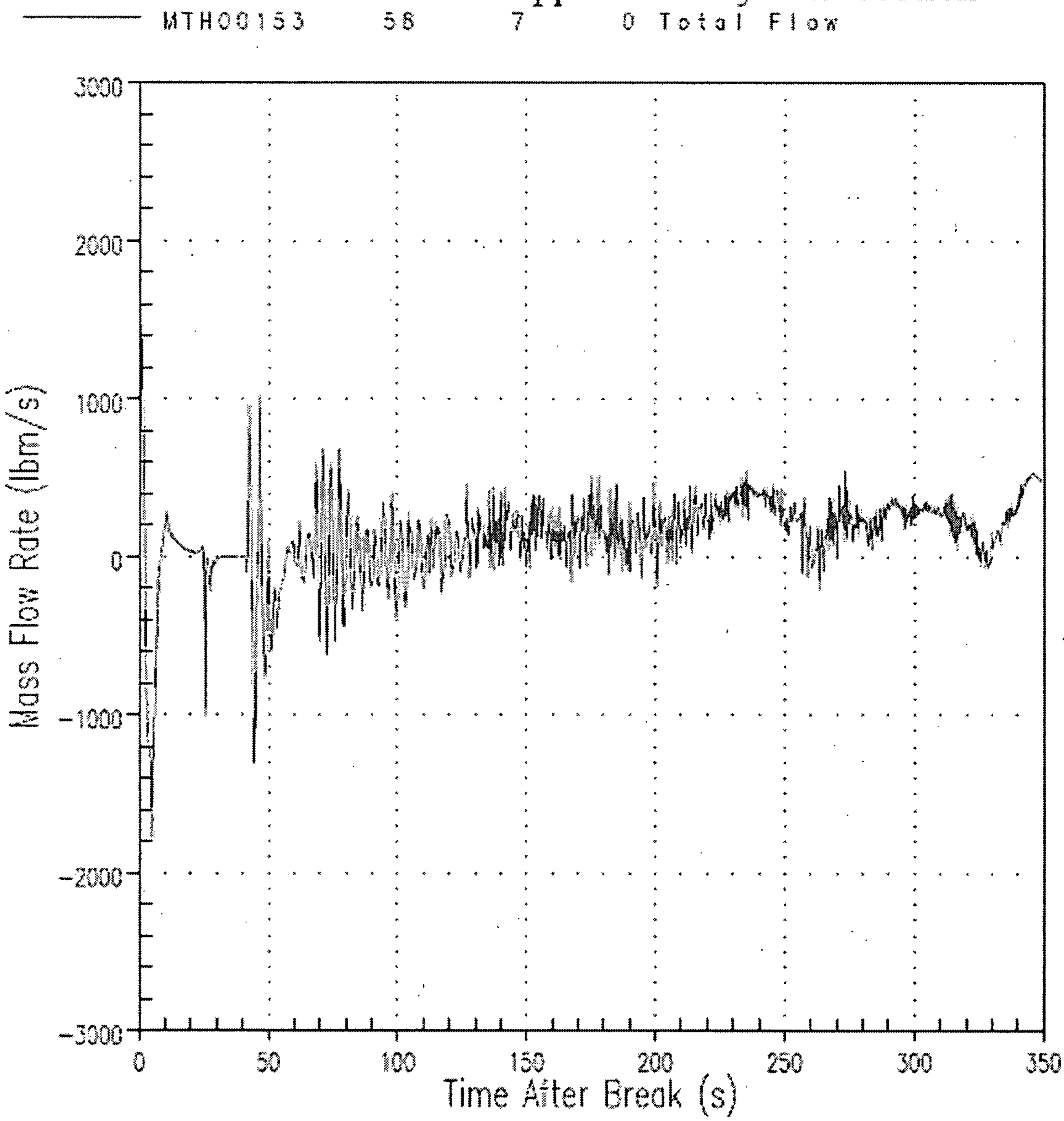
PTN Limiting PCT Case

Figure c19: Total Flow from Intact Loop Downcomer Channel 6 Core – Section 3 – at Approximately Mid-Section



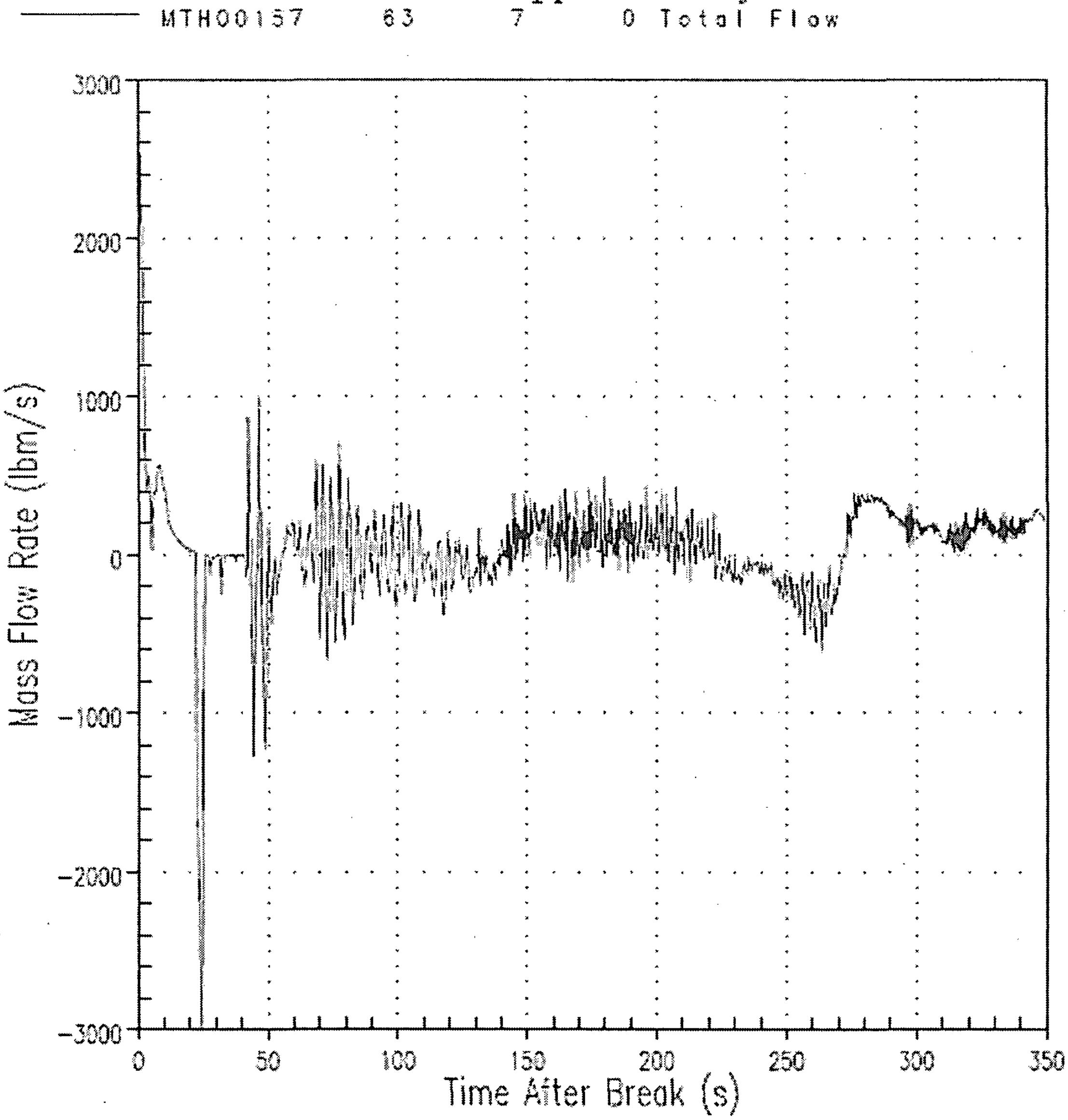
PTN Limiting PCT Case

Figure c20: Total Flow from Intact Loop Downcomer Channel 58 Core – Section 3 – at Approximately Mid-Section



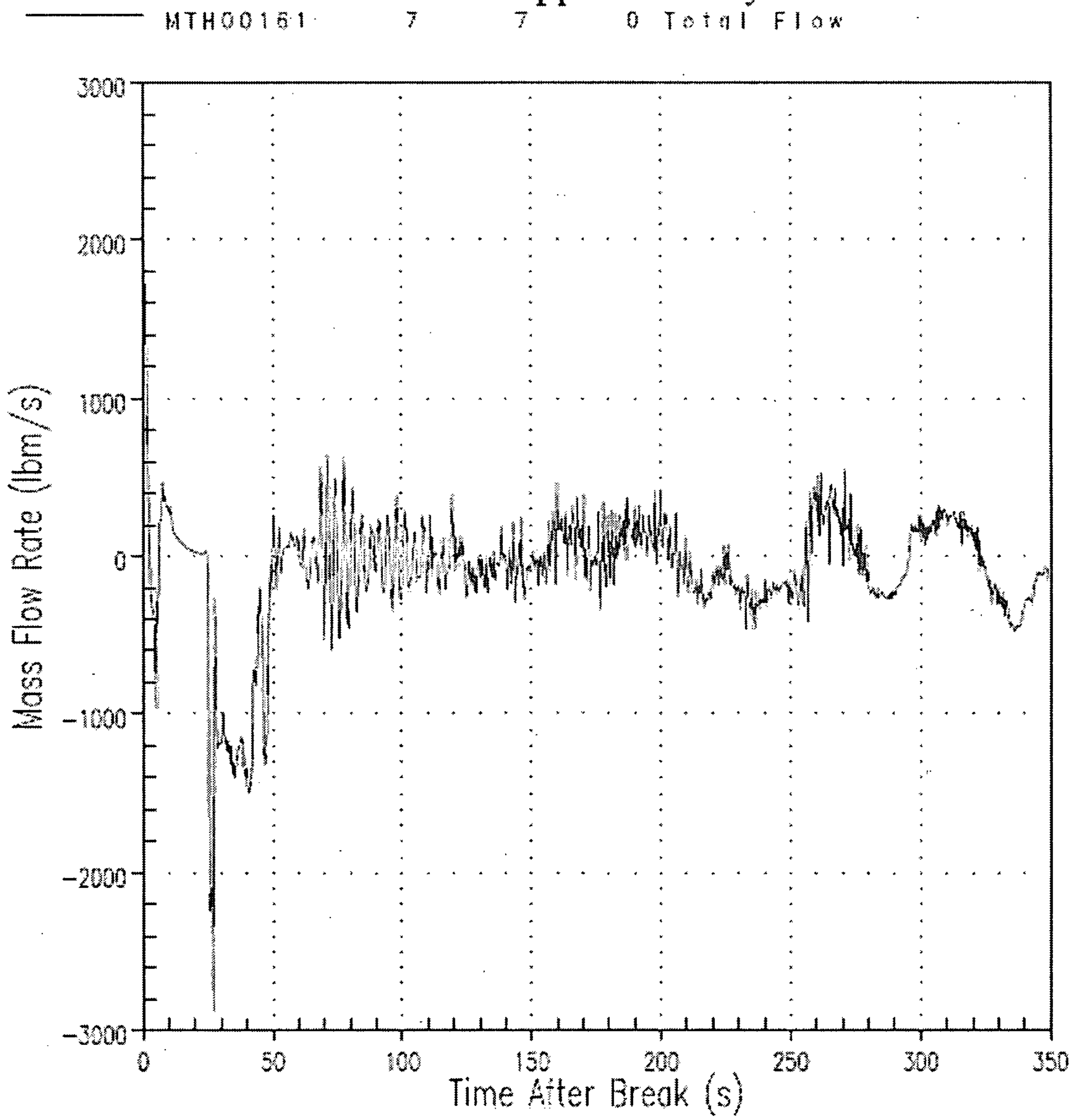
PTN Limiting PCT Case

Figure c21: Total Flow from Intact Loop Downcomer Channel 63 Core – Section 3 – at Approximately Mid-Section



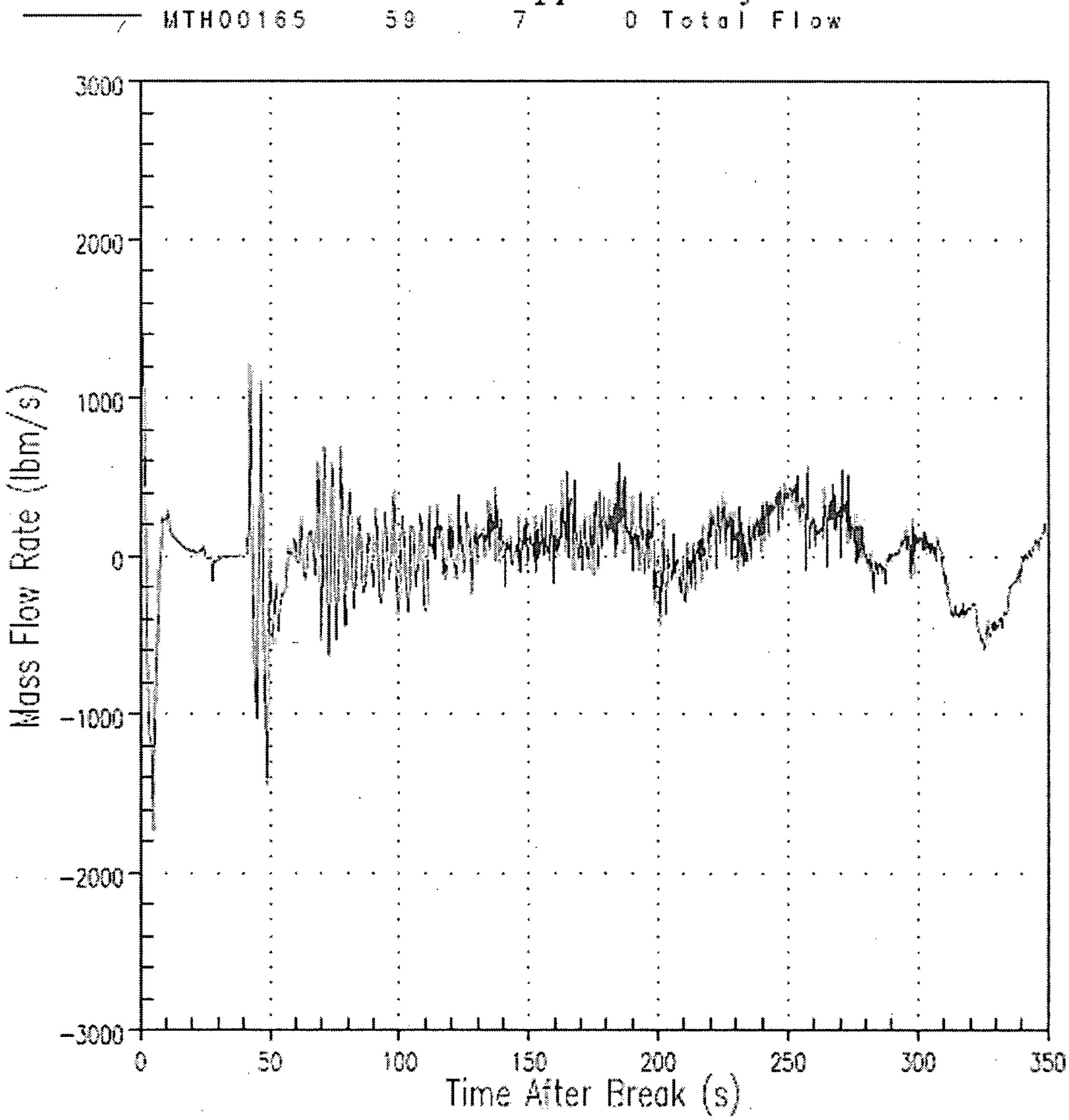
PTN Limiting PCT Case

Figure c22: Total Flow from Intact Loop Downcomer Channel 7 Core – Section 3 – at Approximately Mid-Section



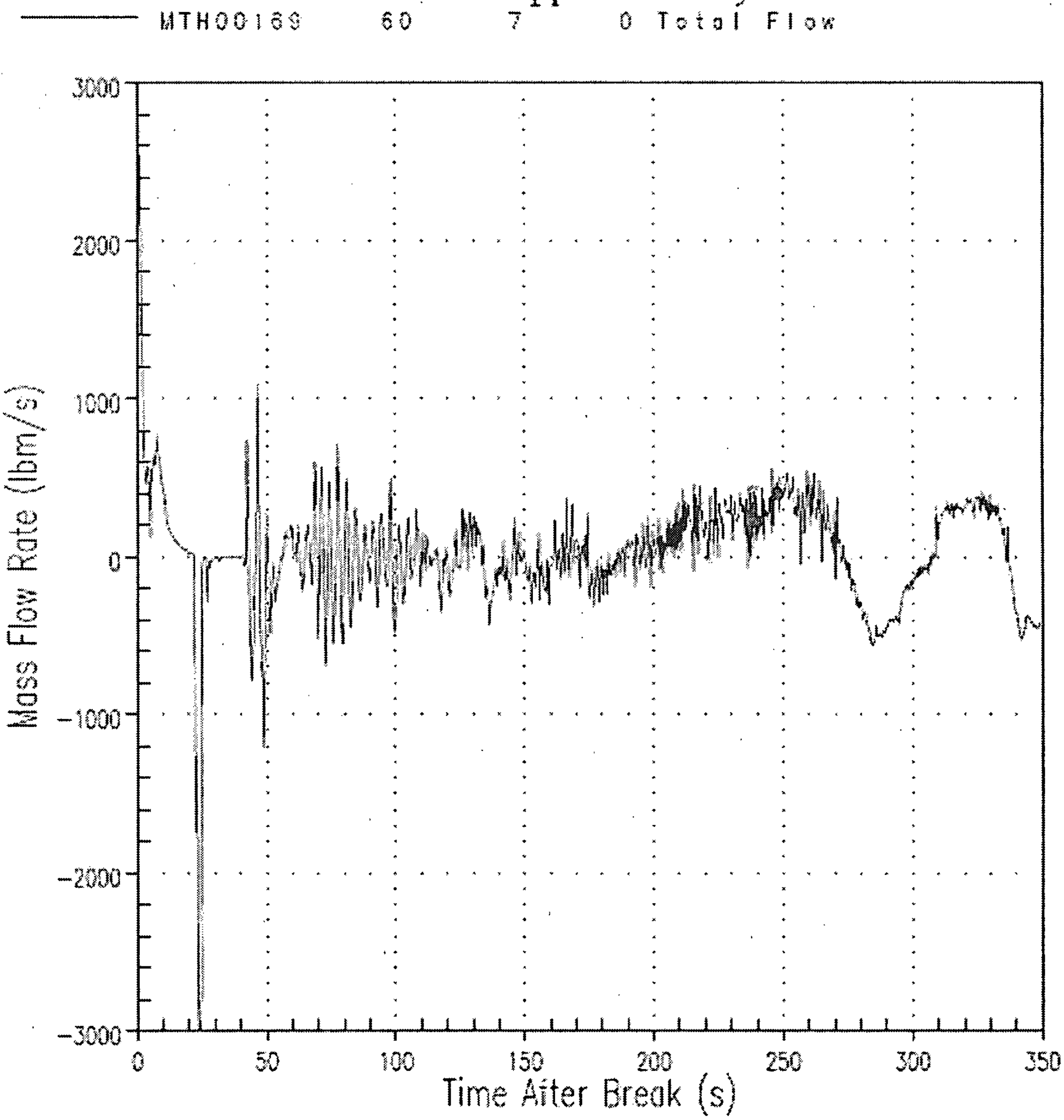
PTN Limiting PCT Case

Figure c23: Total Flow from Intact Loop Downcomer Channel 59 Core – Section 3 – at Approximately Mid-Section



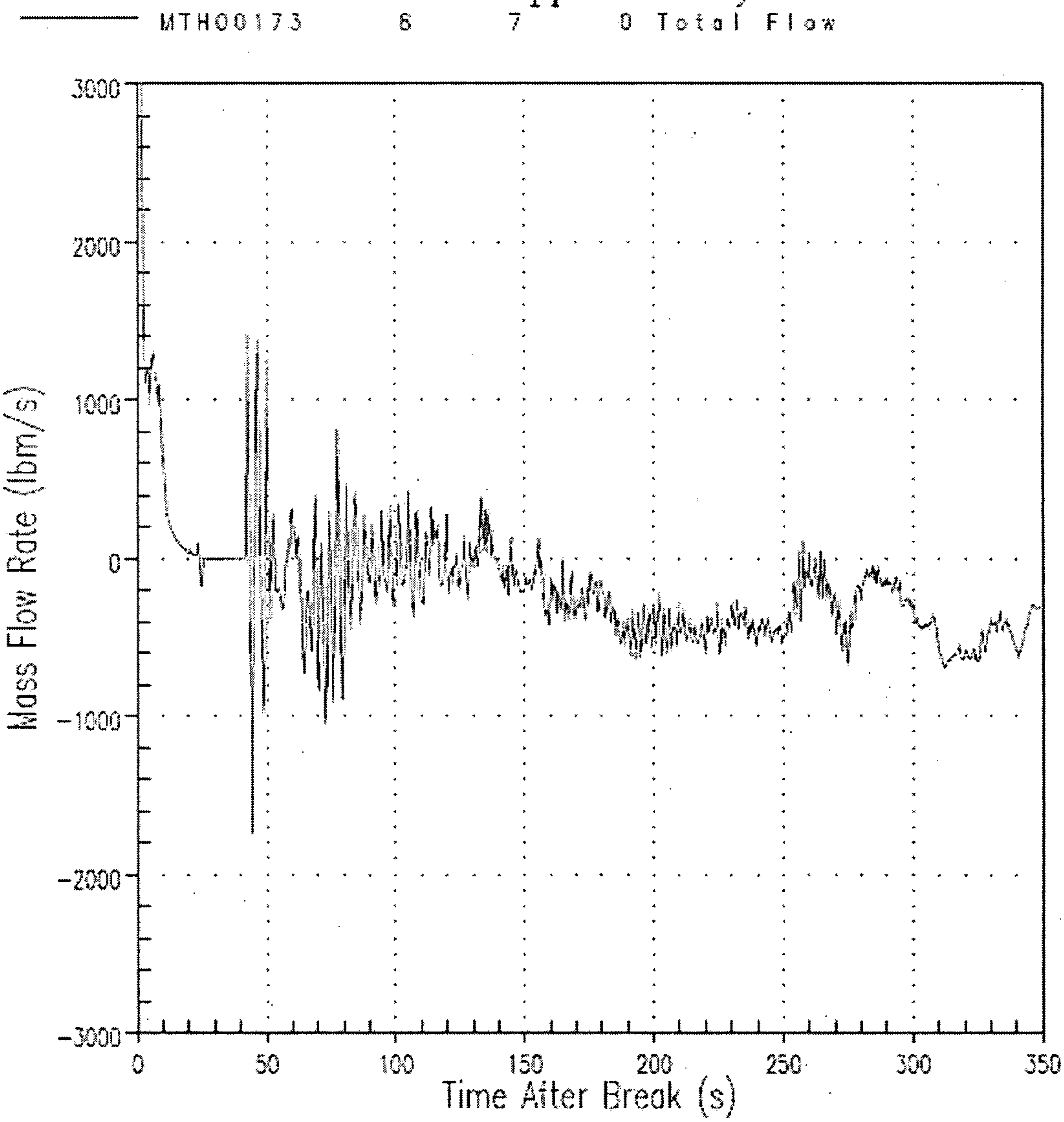
PTN Limiting PCT Case

Figure c24: Total Flow from Intact Loop Downcomer Channel 60
Core – Section 3 – at Approximately Mid-Section



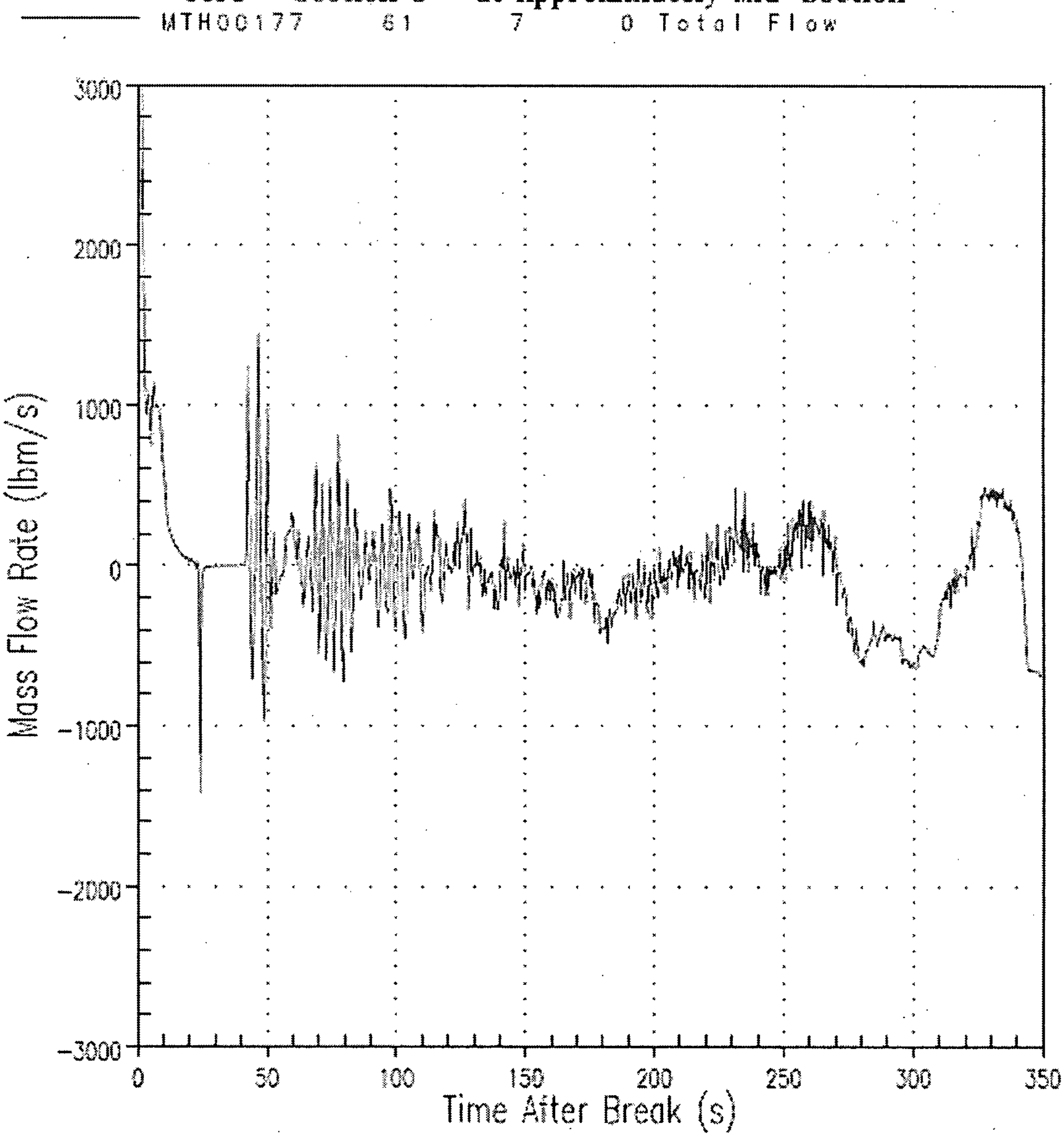
PTN Limiting PCT Case

Figure c25: Total Flow from Broken Loop Downcomer Channel 8 Core – Section 3 – at Approximately Mid-Section



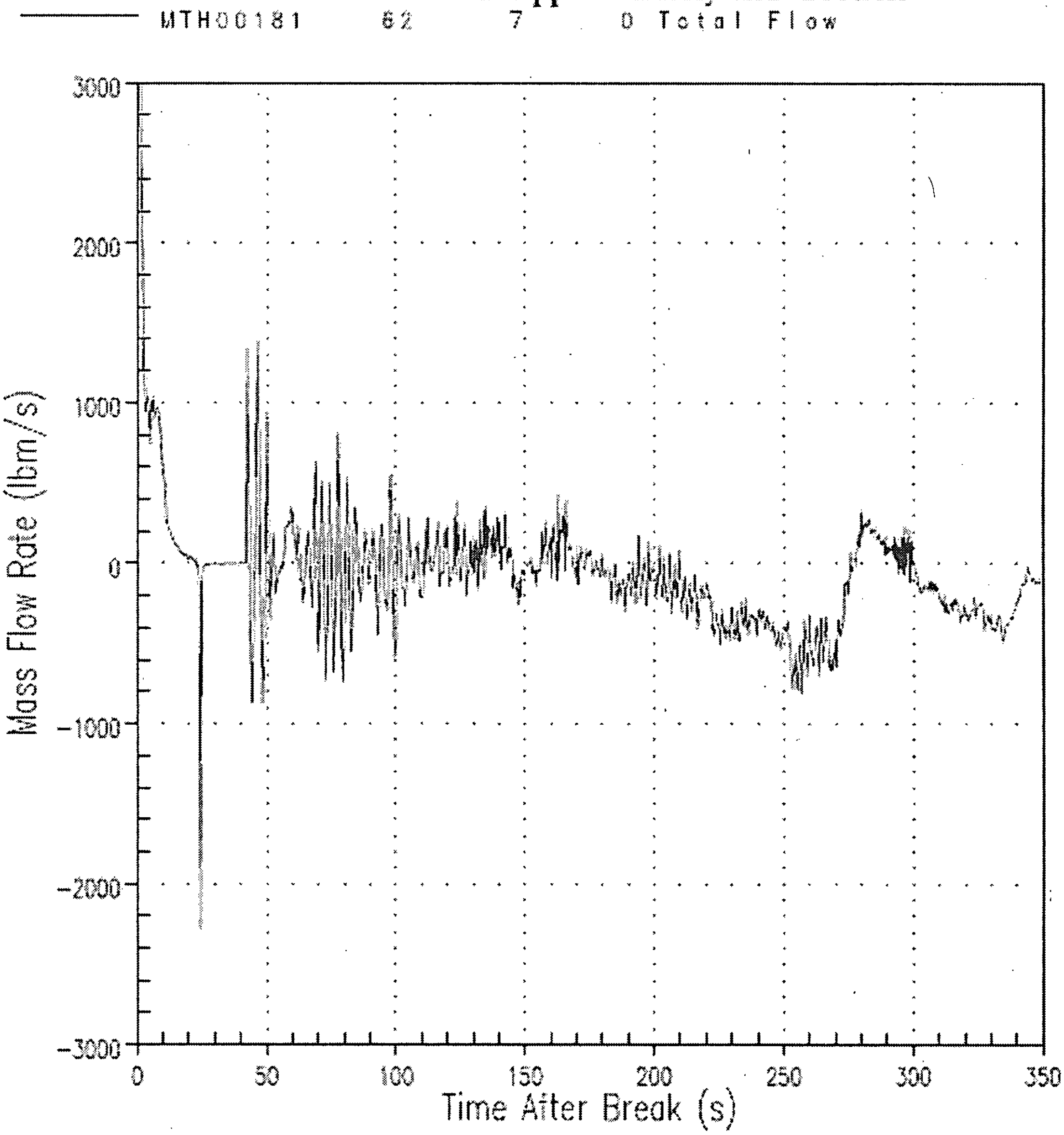
PTN Limiting PCT Case

Figure c26: Total Flow from Adjacen to Broken Loop Downcomer Channel 61 Core – Section 3 – at Approximately Mid-Section



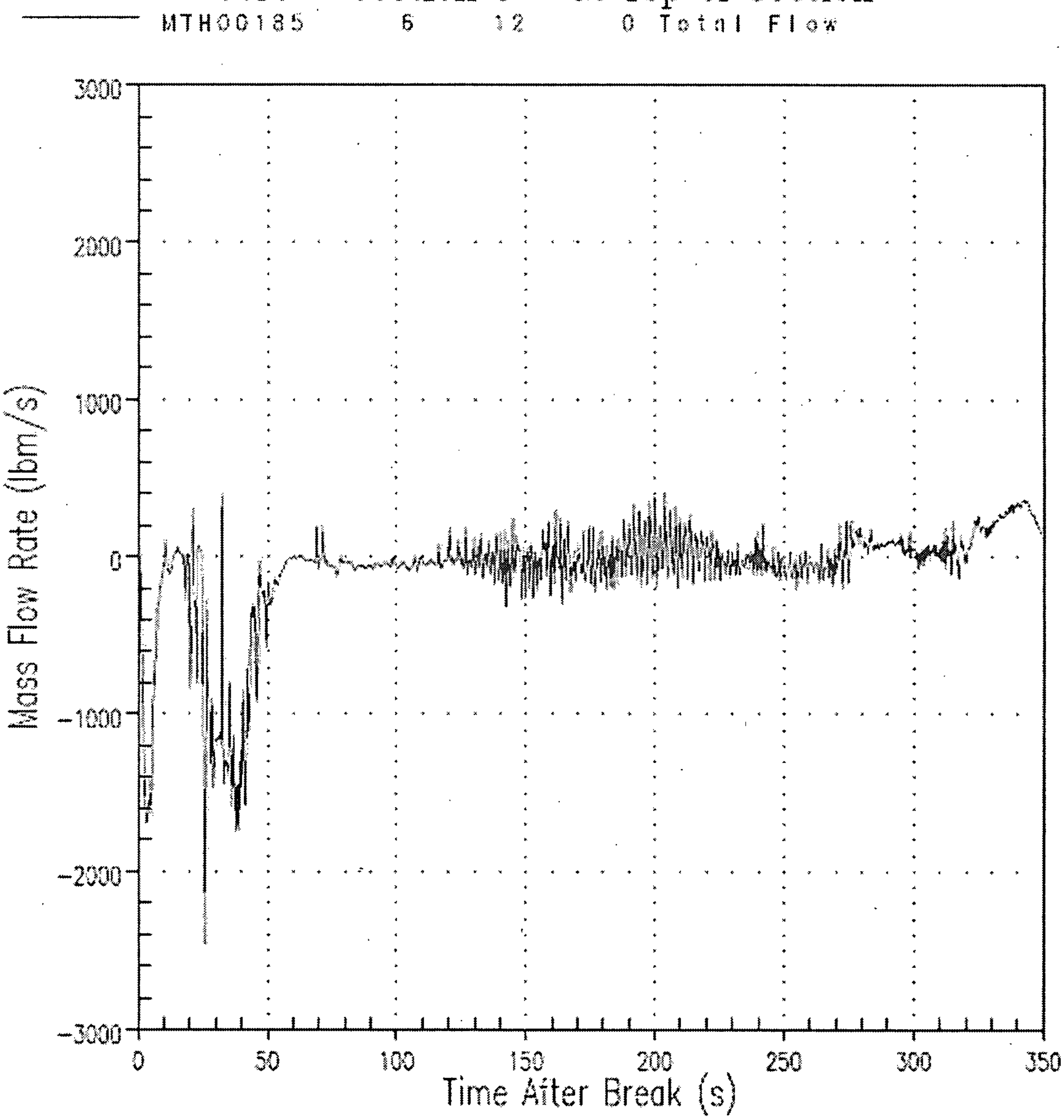
PTN Limiting PCT Case

Figure c27: Total Flow from Adjacen to Broken Loop Downcomer Channel 62 Core - Section 3 - at Approximately Mid-Section



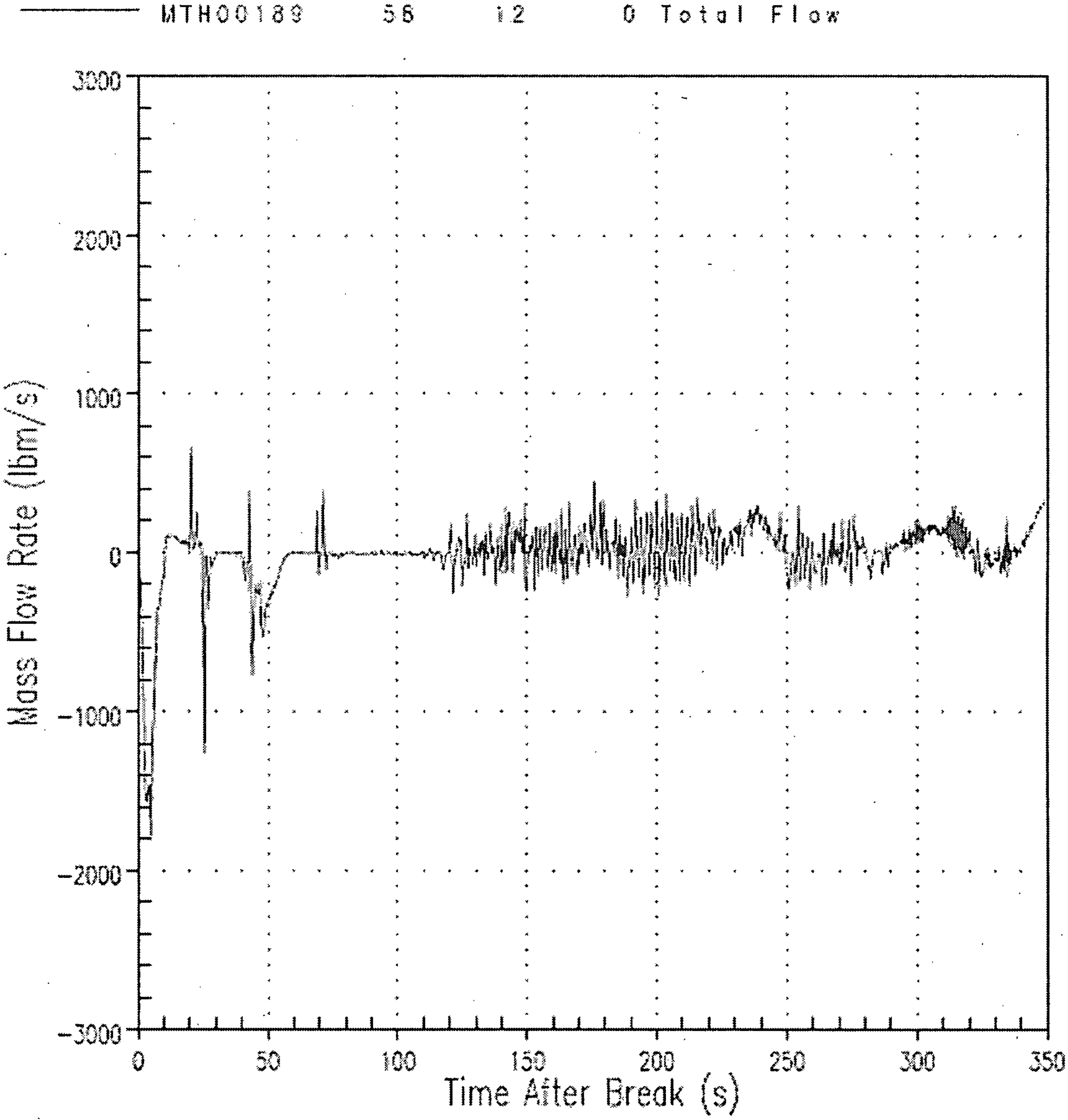
PTN Limiting PCT Case

Figure c28: Total Flow from Intact Loop Downcomer Channel 6 Core – Section 3 – at Top of Section



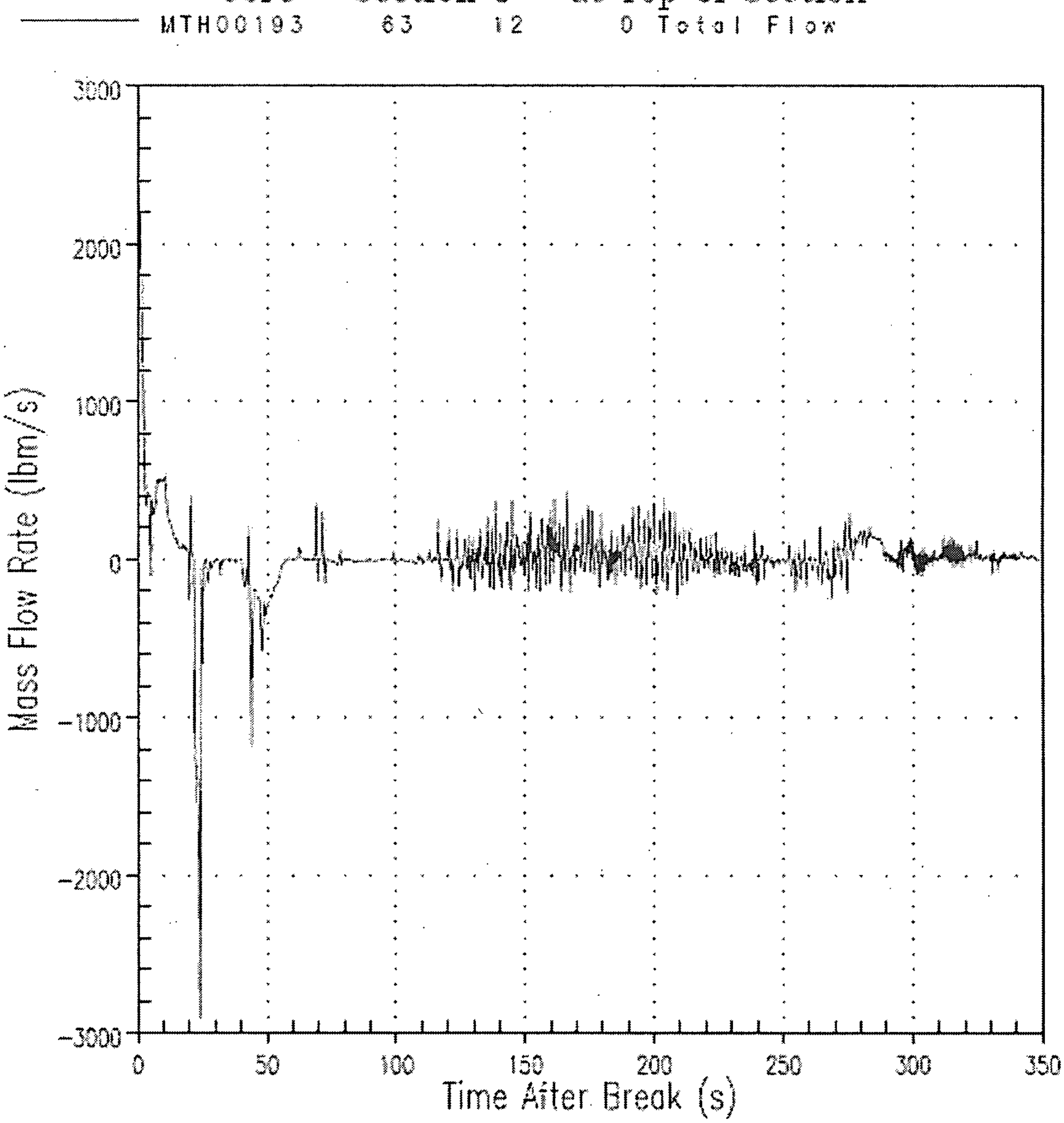
PTN Limiting PCT Case

Figure c29: Total Flow from Intact Loop Downcomer Channel 58
Core - Section 3 - at Top of Section



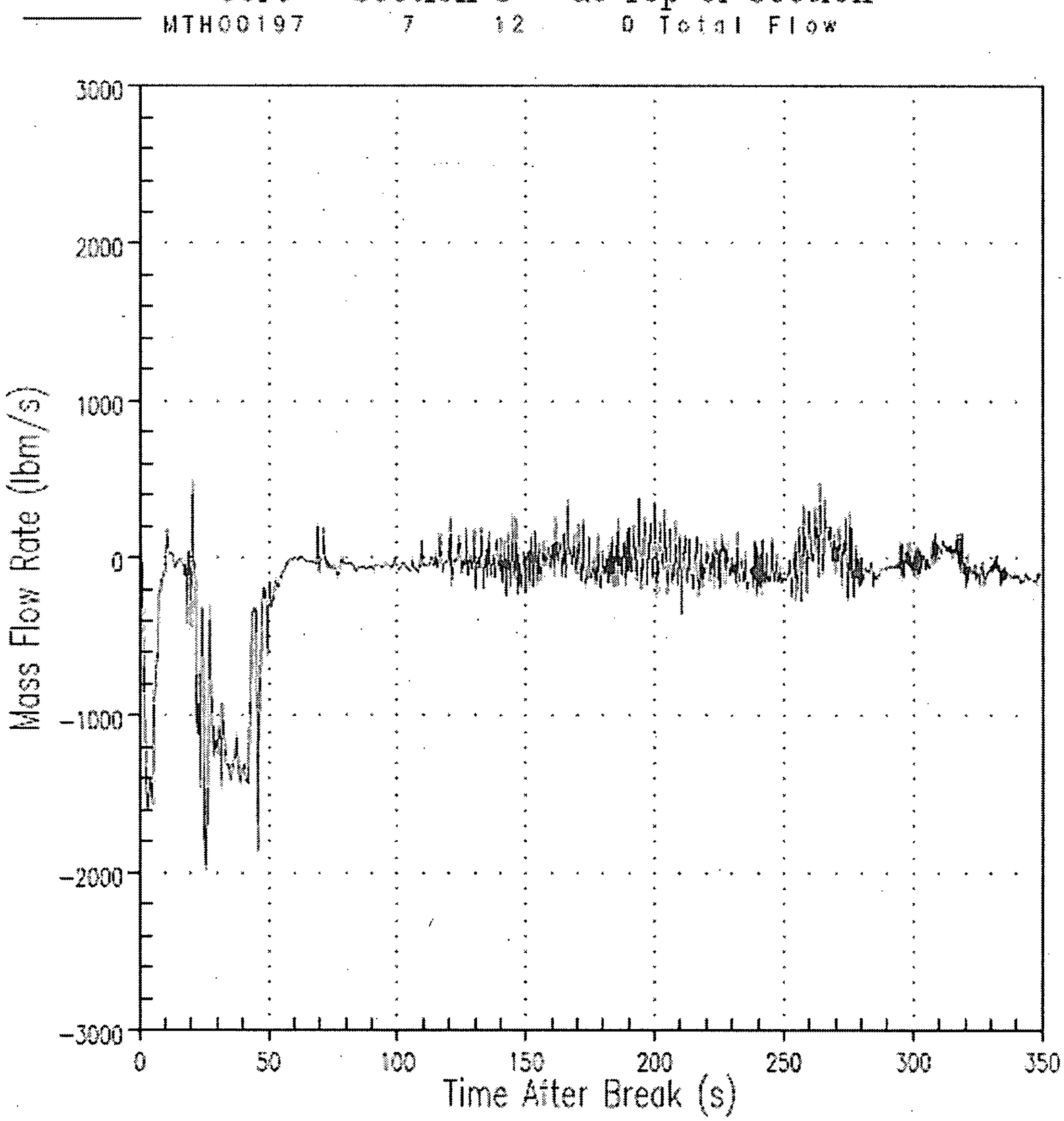
PTN Limiting PCT Case

Figure c30: Total Flow from Intact Loop Downcomer Channel 63 Core - Section 3 - at Top of Section



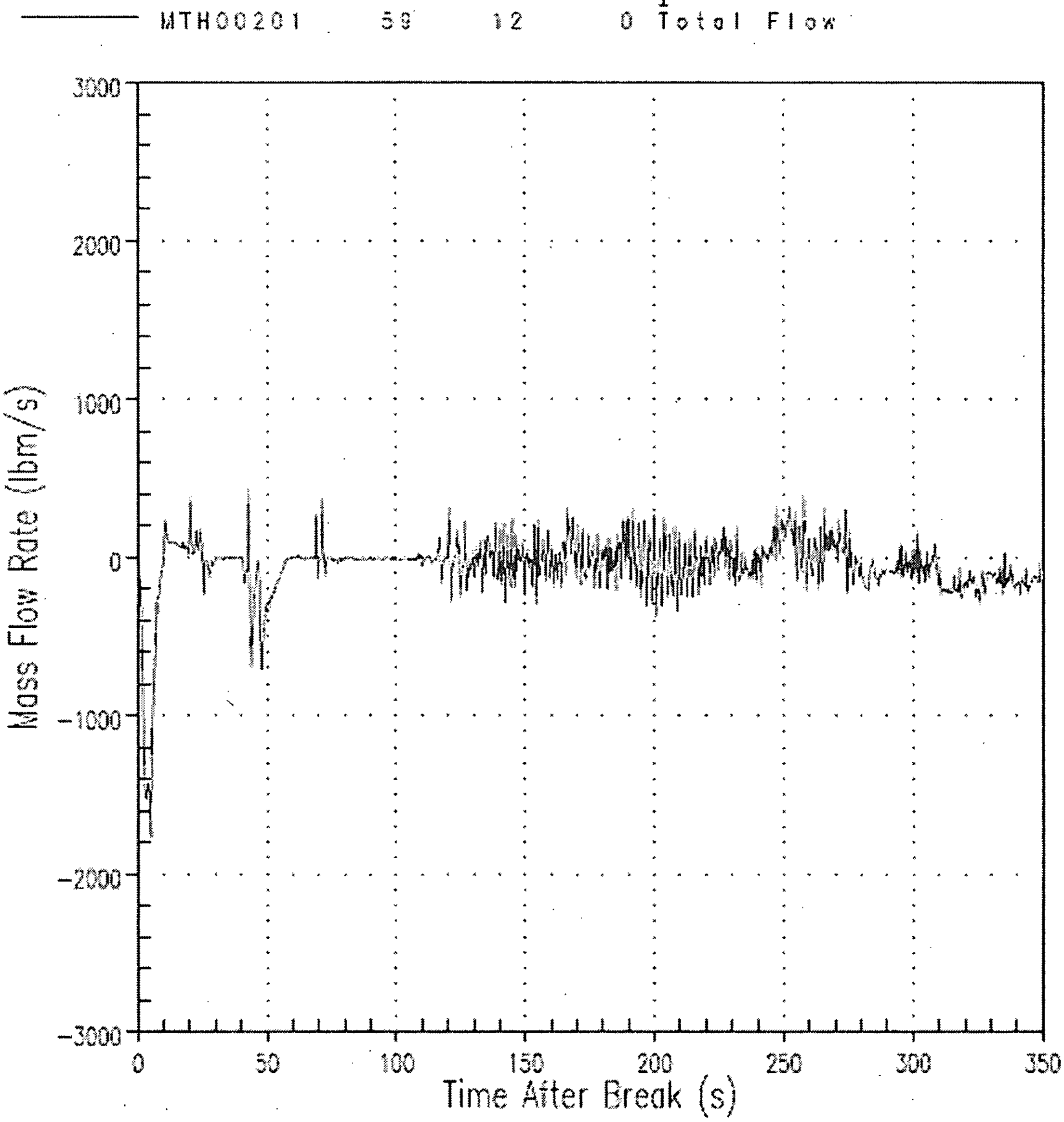
PTN Limiting PCT Case

Figure c31: Total Flow from Intact Loop Downcomer Channel 7 Core – Section 3 – at Top of Section



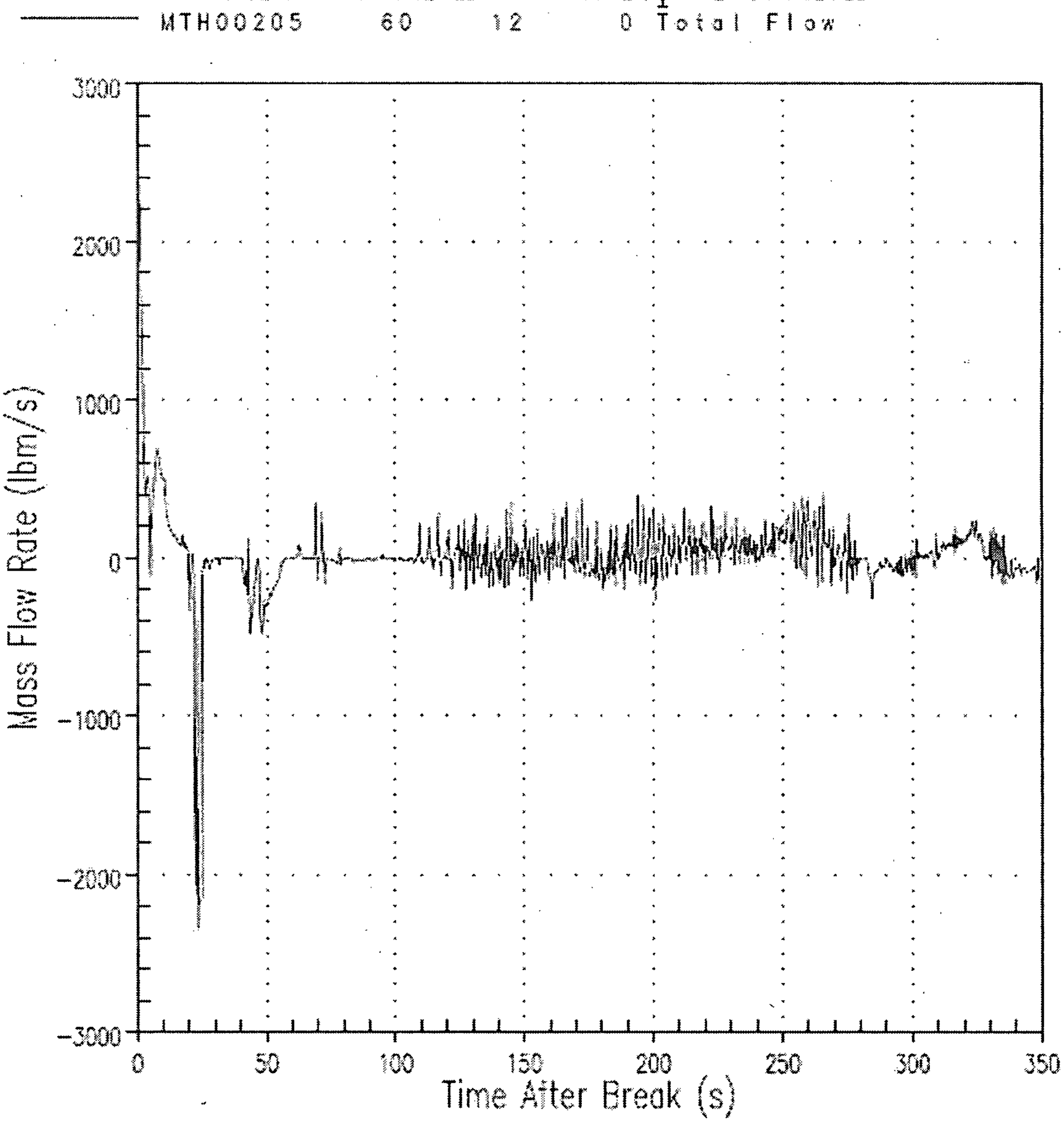
PTN Limiting PCT Case

Figure c32: Total Flow from Intact Loop Downcomer Channel 59 Core – Section 3 – at Top of Section



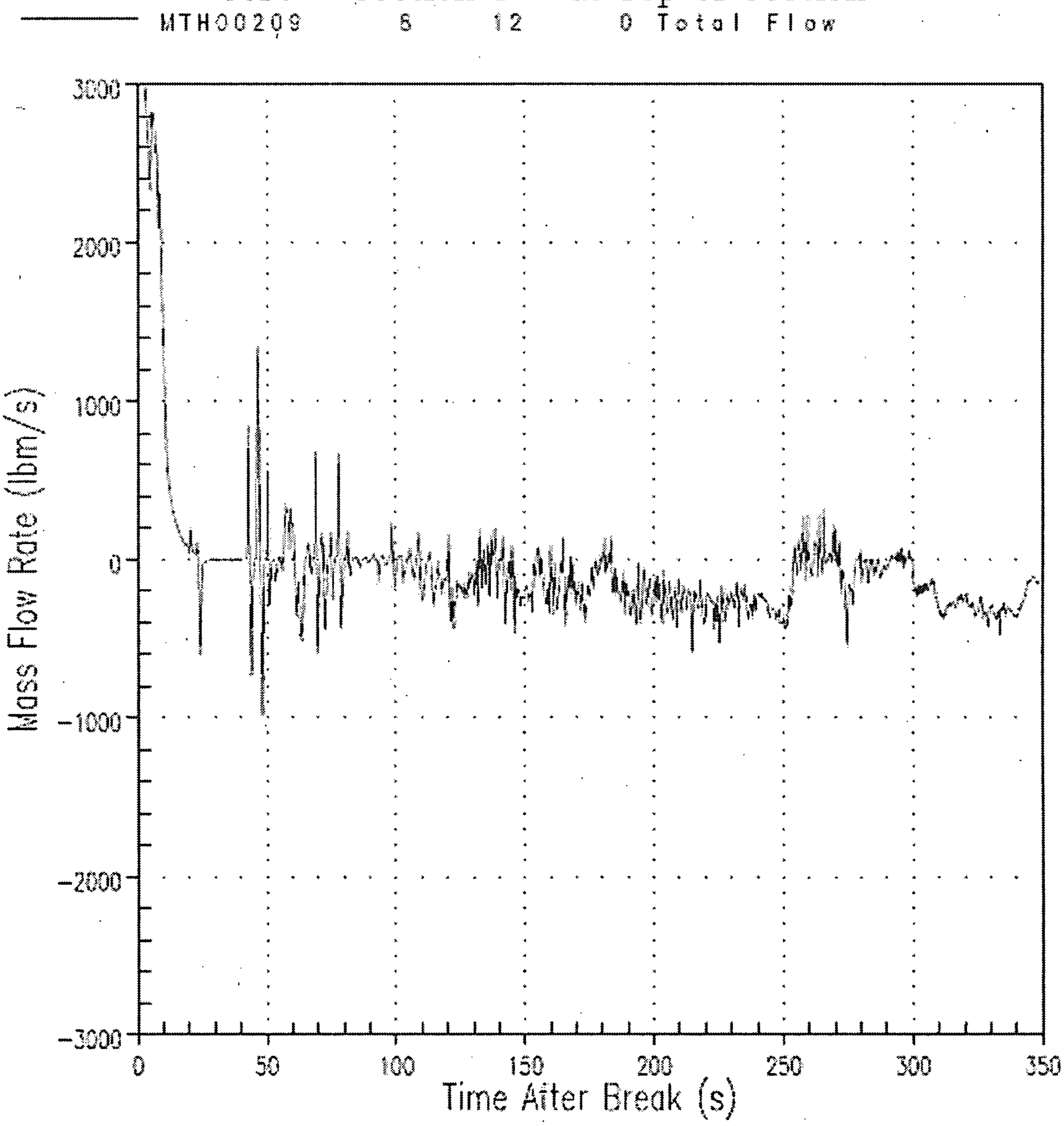
PTN Limiting PCT Case

Figure c33: Total Flow from Intact Loop Downcomer Channel 60
Core – Section 3 – at Top of Section



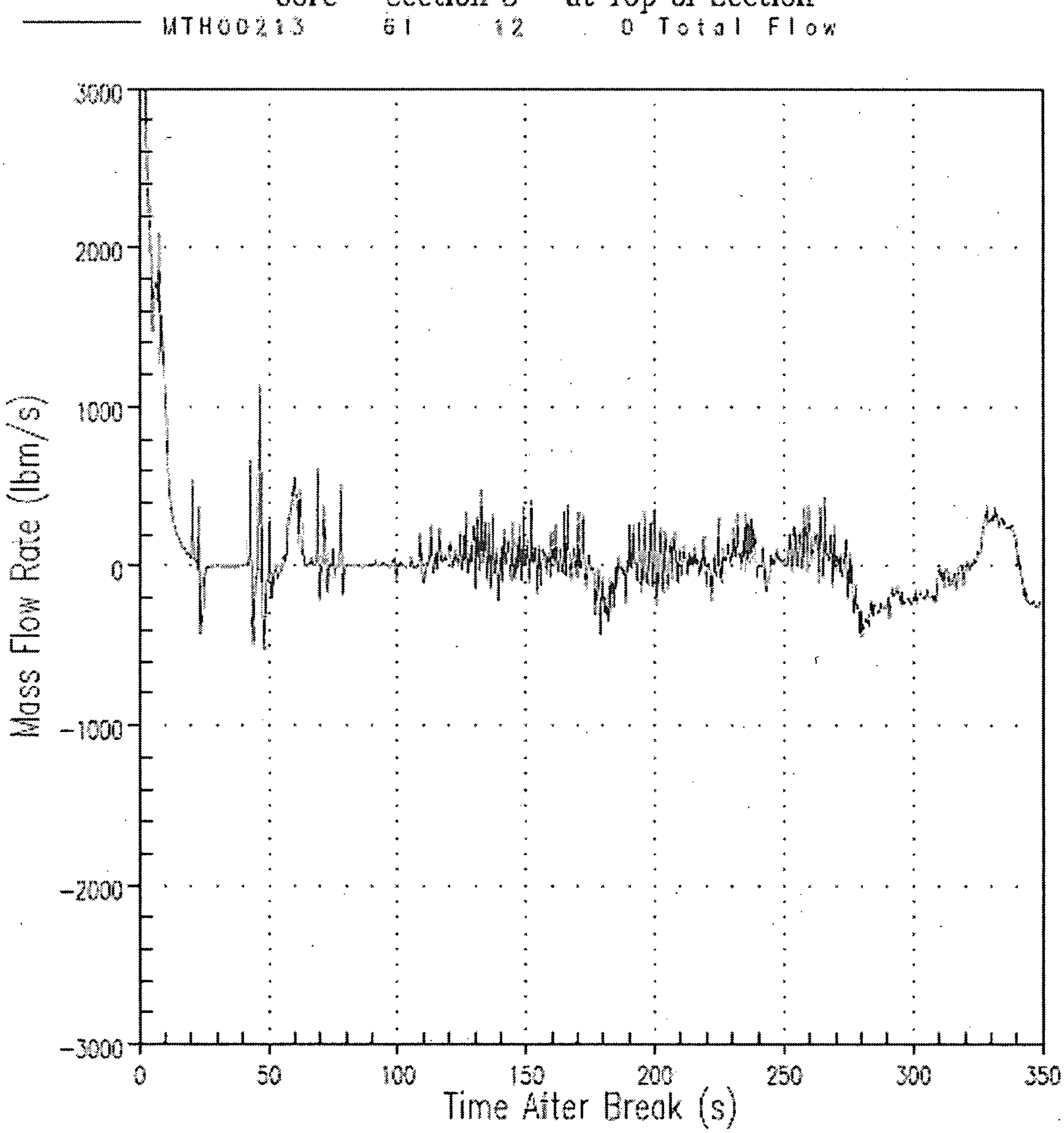
PTN Limiting PCT Case

Figure c34: Total Flow from Broken Loop Downcomer Channel 8 Core – Section 3 – at Top of Section



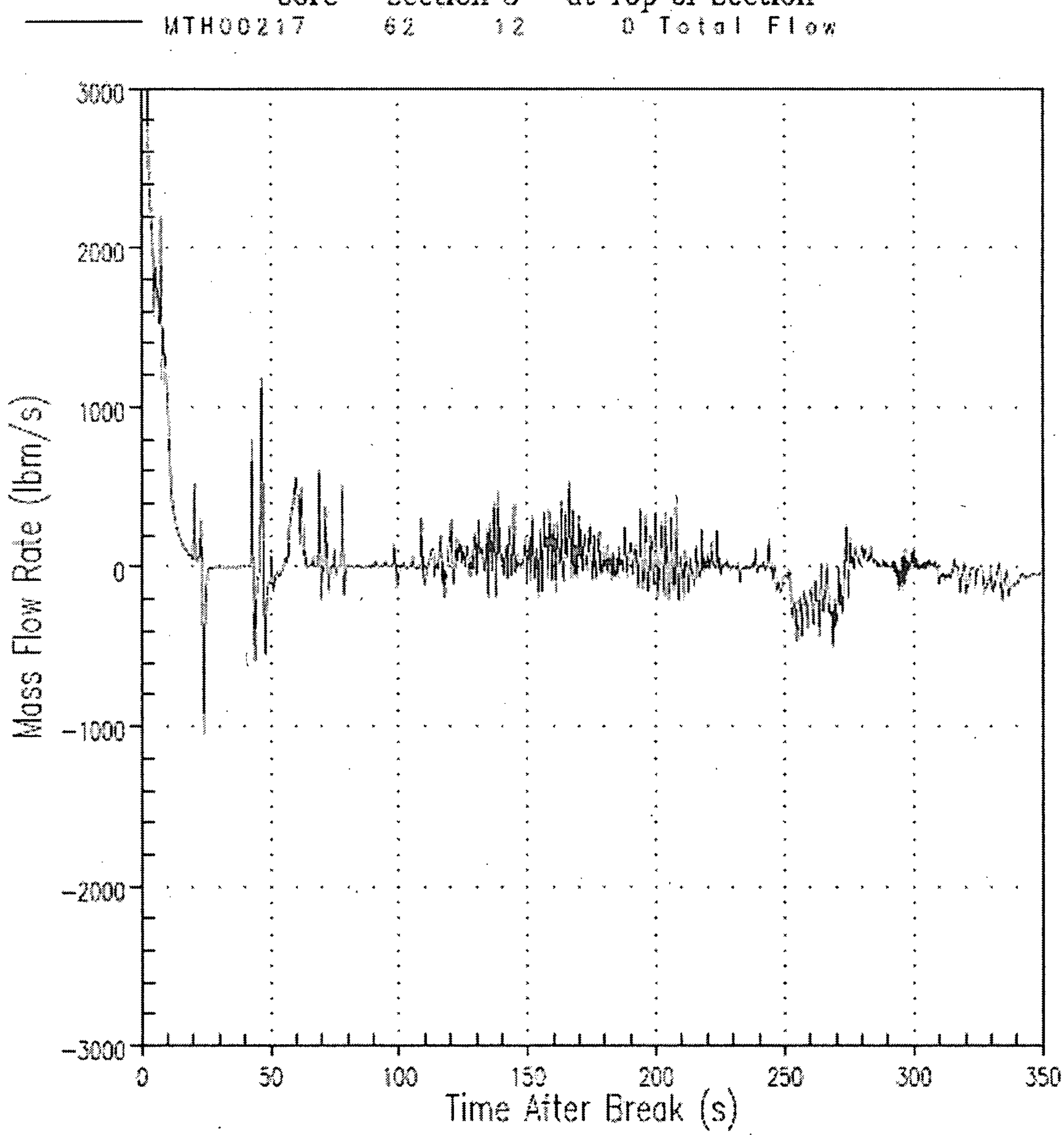
PTN Limiting PCT Case

Figure c35: Total Flow from Adjacen to Broken Loop Downcomer Channel 61 Core – Section 3 – at Top of Section



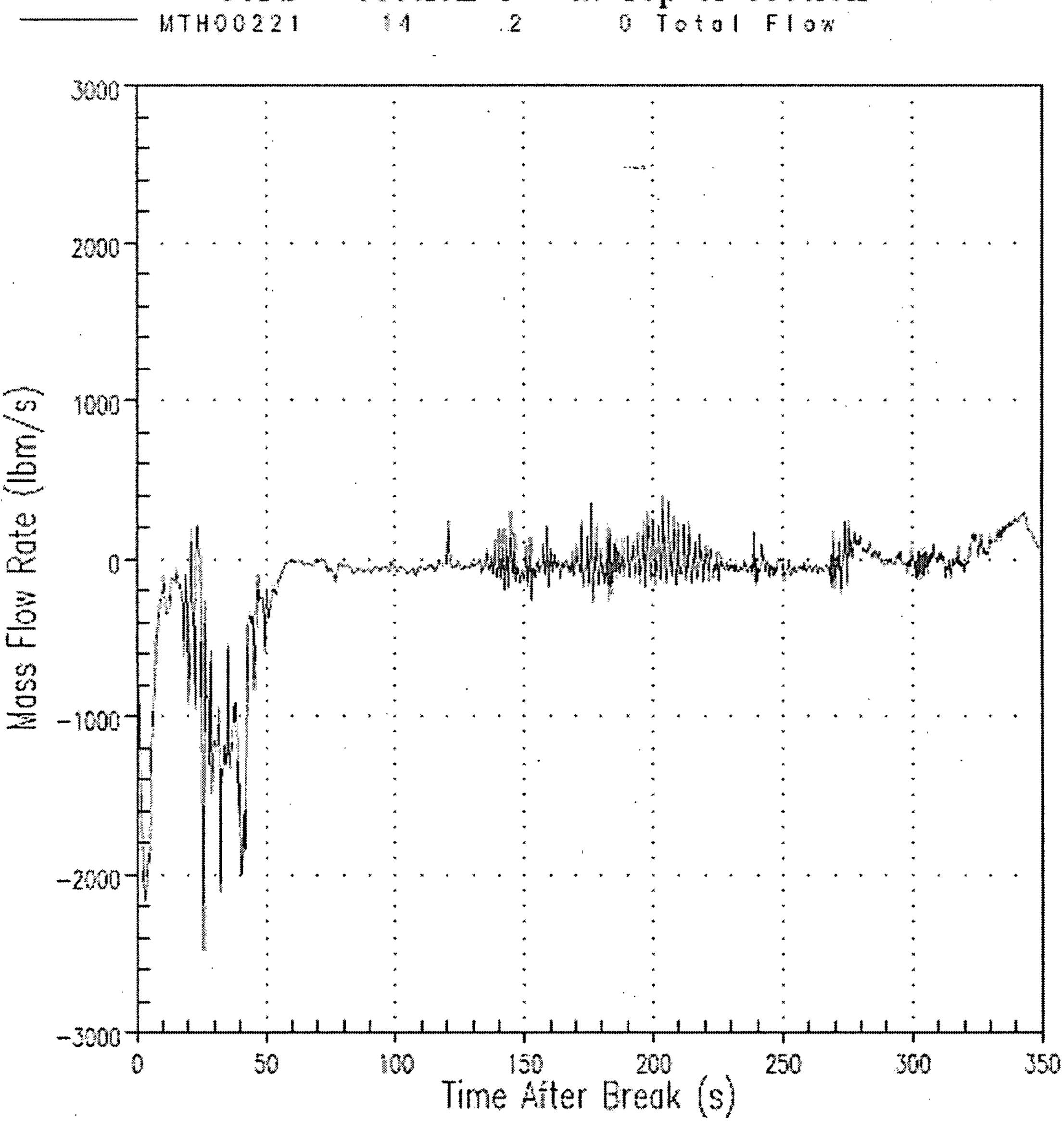
PTN Limiting PCT Case

Figure e36: Total Flow from Adjacen to Broken Loop Downcomer Channel 62 Core – Section 3 – at Top of Section



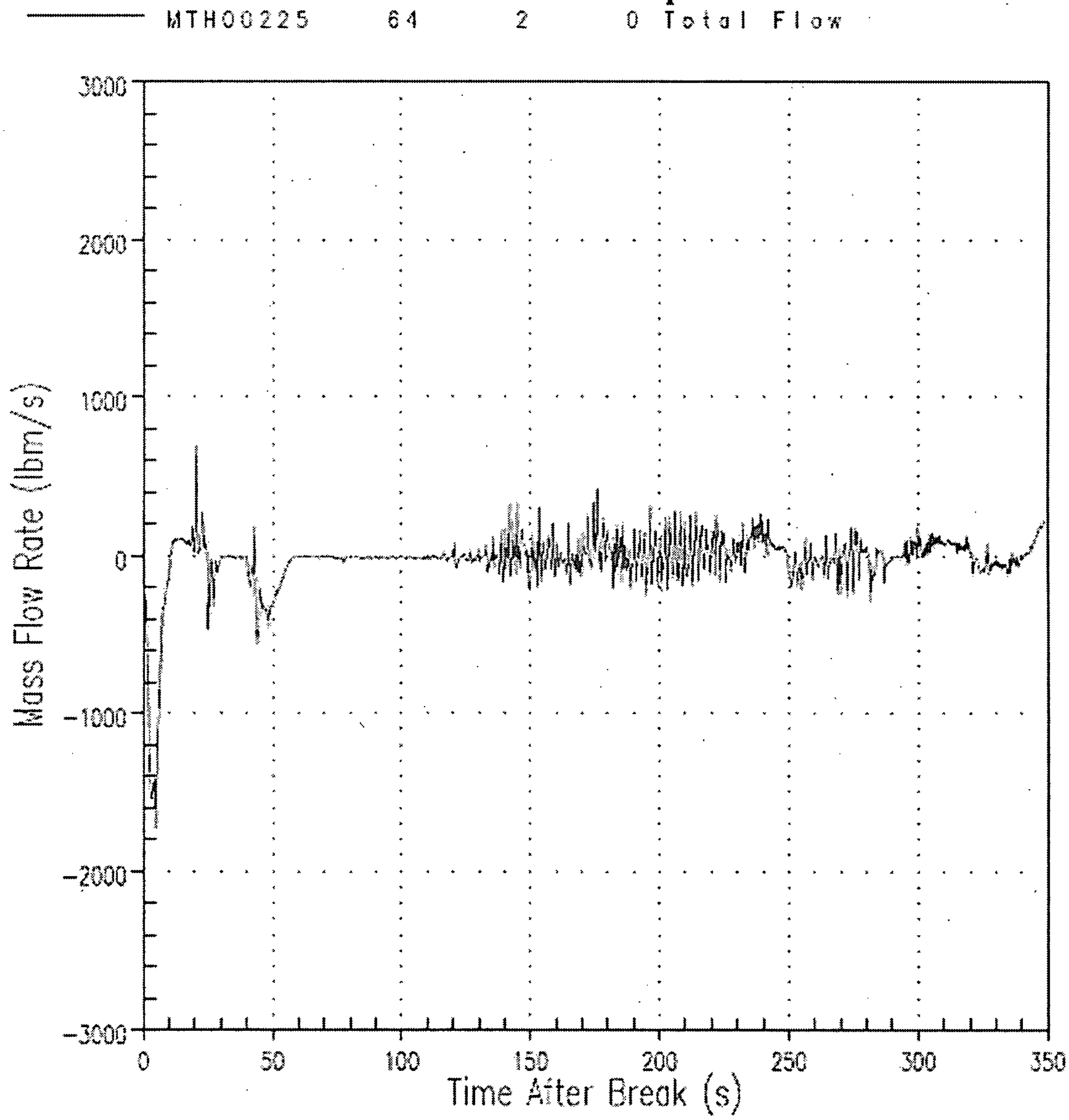
PTN Limiting PCT Case

Figure c37: Total Flow from Intact Loop Downcomer Channel 14
CCFL – Section 4 – at Top of Section



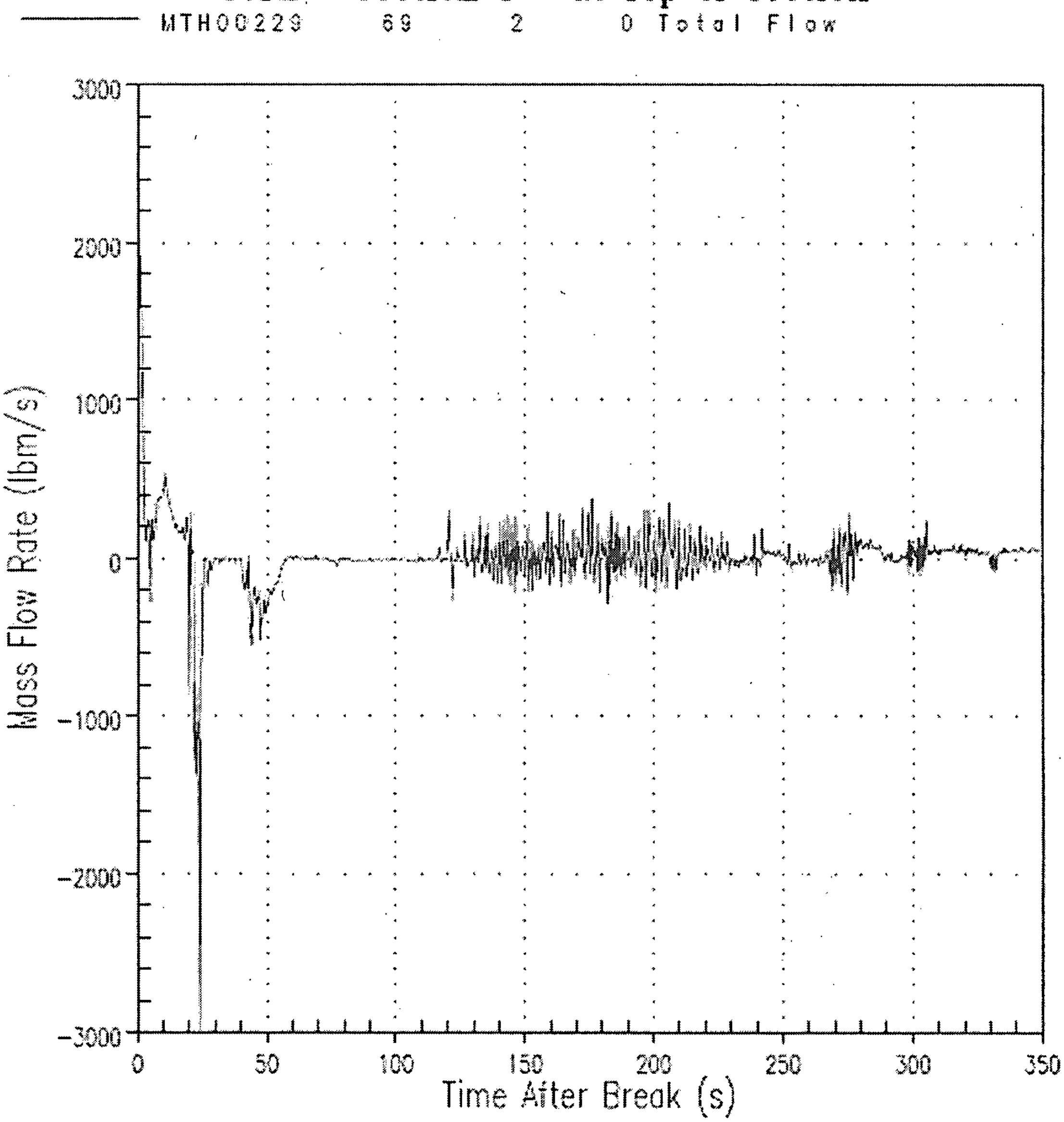
PTN Limiting PCT Case

Figure c38: Total Flow from Intact Loop Downcomer Channel 64
CCFL – Section 4 – at Top of Section



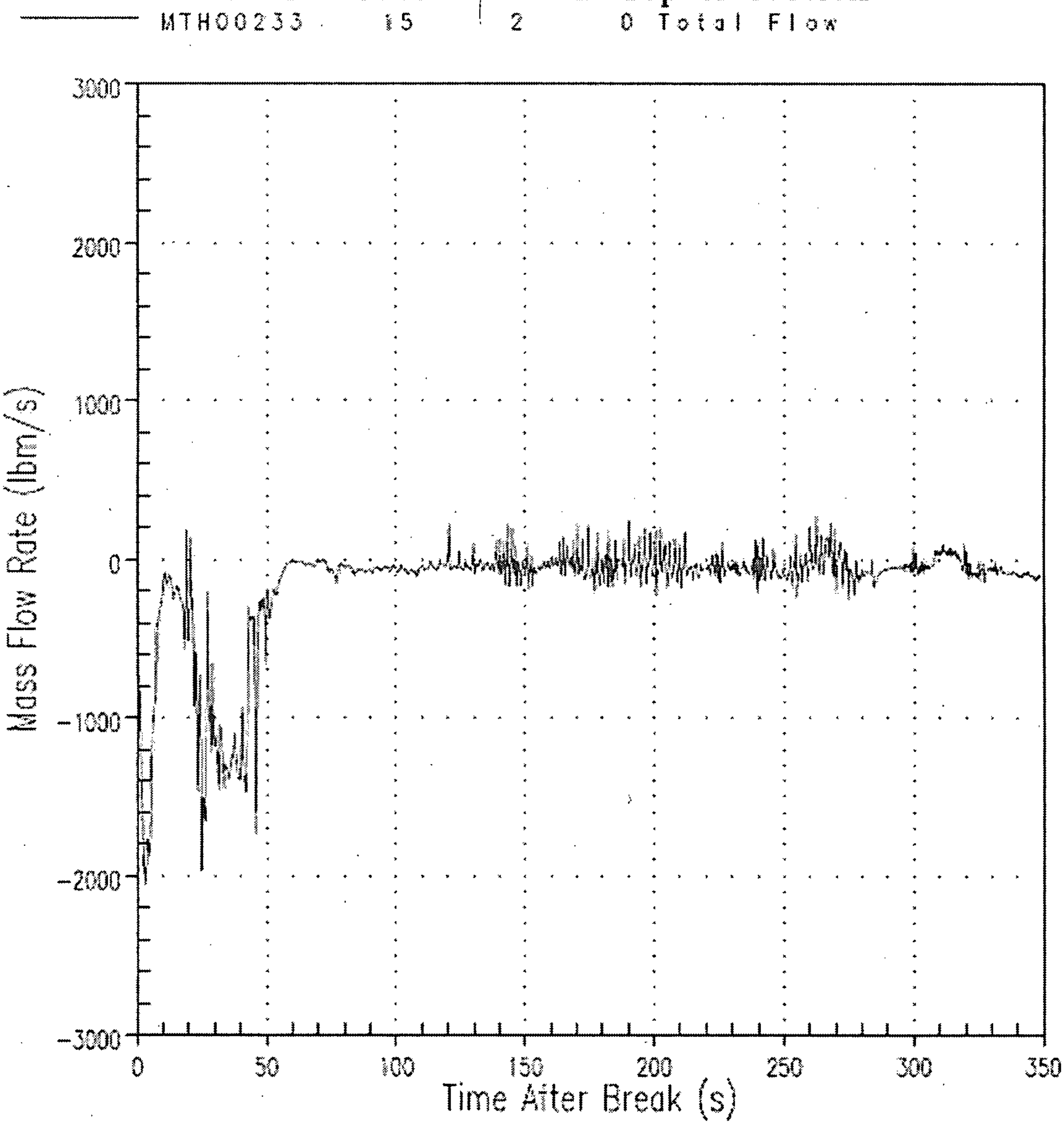
PTN Limiting PCT Case

Figure c39: Total Flow from Intact Loop Downcomer Channel 69
CCFL – Section 4 – at Top of Section



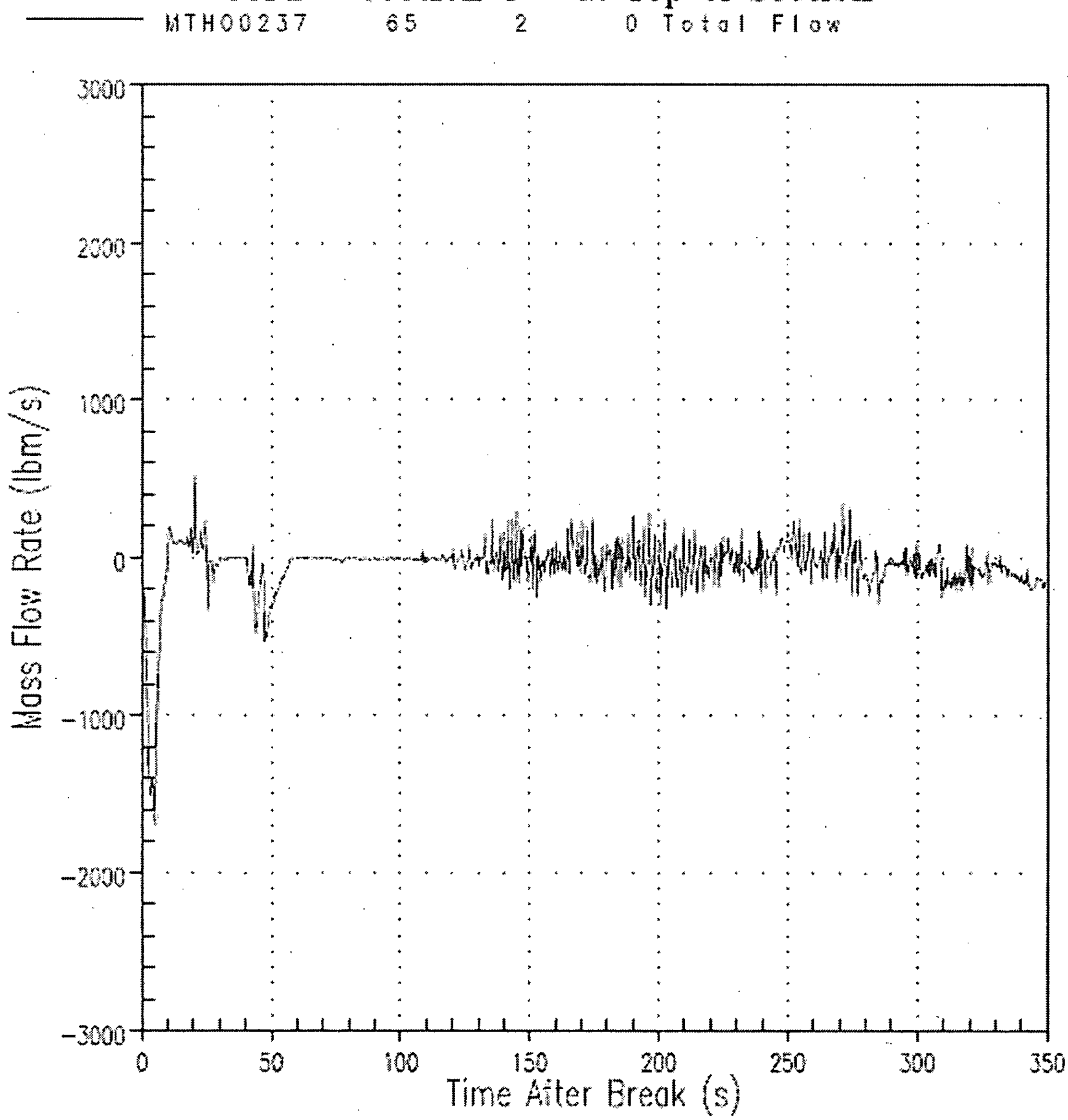
PTN Limiting PCT Case

Figure c40: Total Flow from Intact Loop Downcomer Channel 15 CCFL – Section 4 – at Top of Section



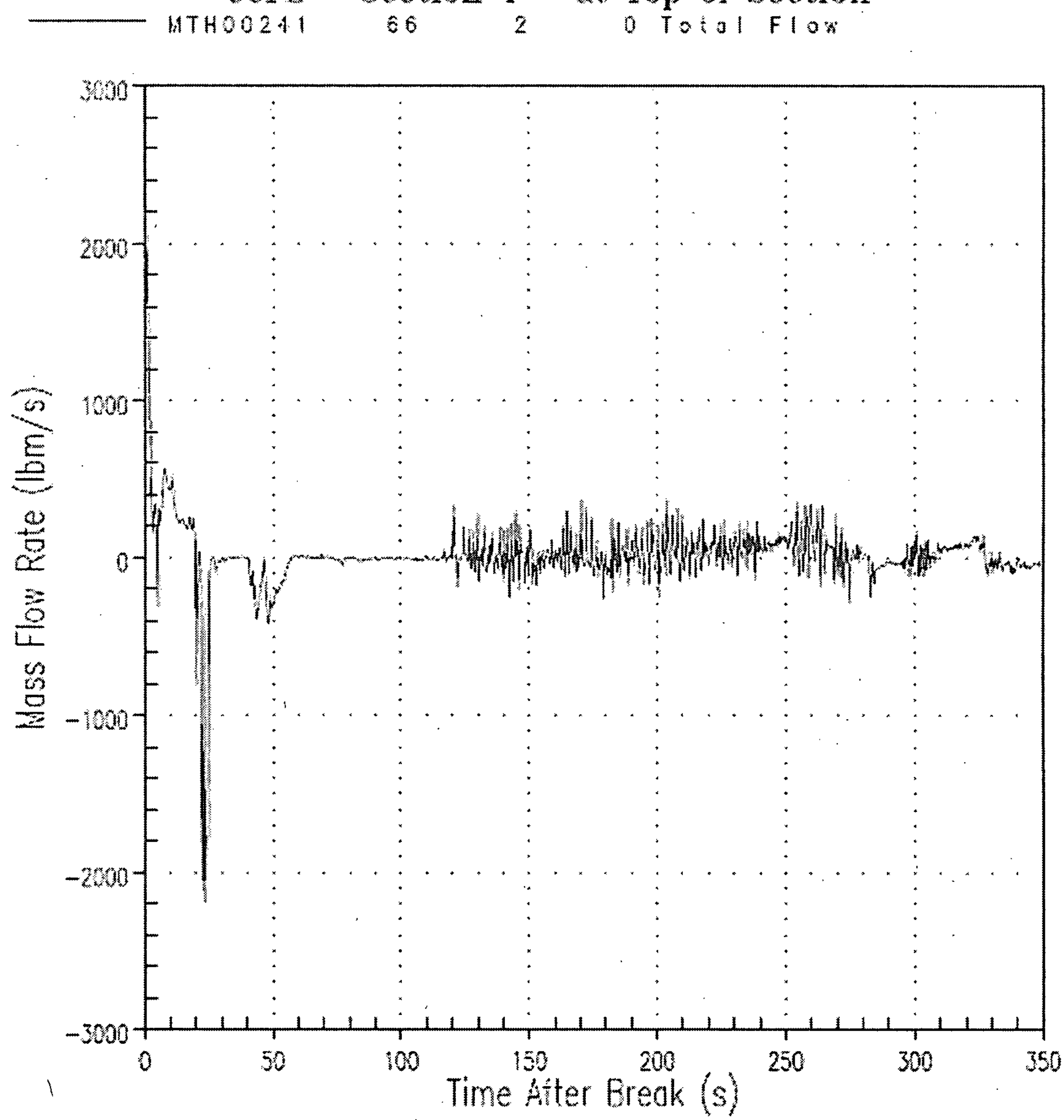
PTN Limiting PCT Case

Figure c41: Total Flow from Intact Loop Downcomer Channel 65
CCFL – Section 4 – at Top of Section



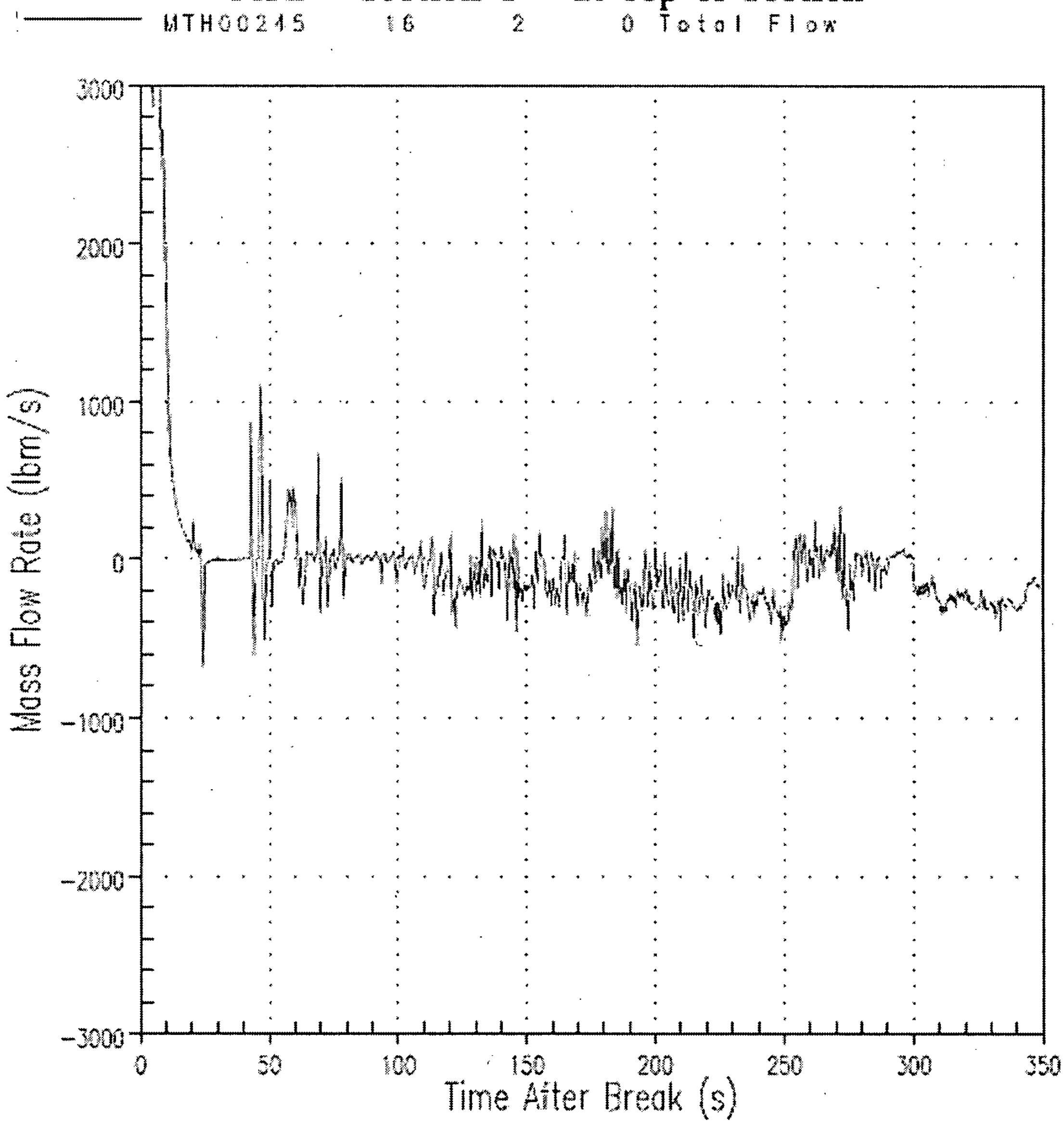
PTN Limiting PCT Case

Figure c42: Total Flow from Intact Loop Downcomer Channel 66
CCFL – Section 4 – at Top of Section



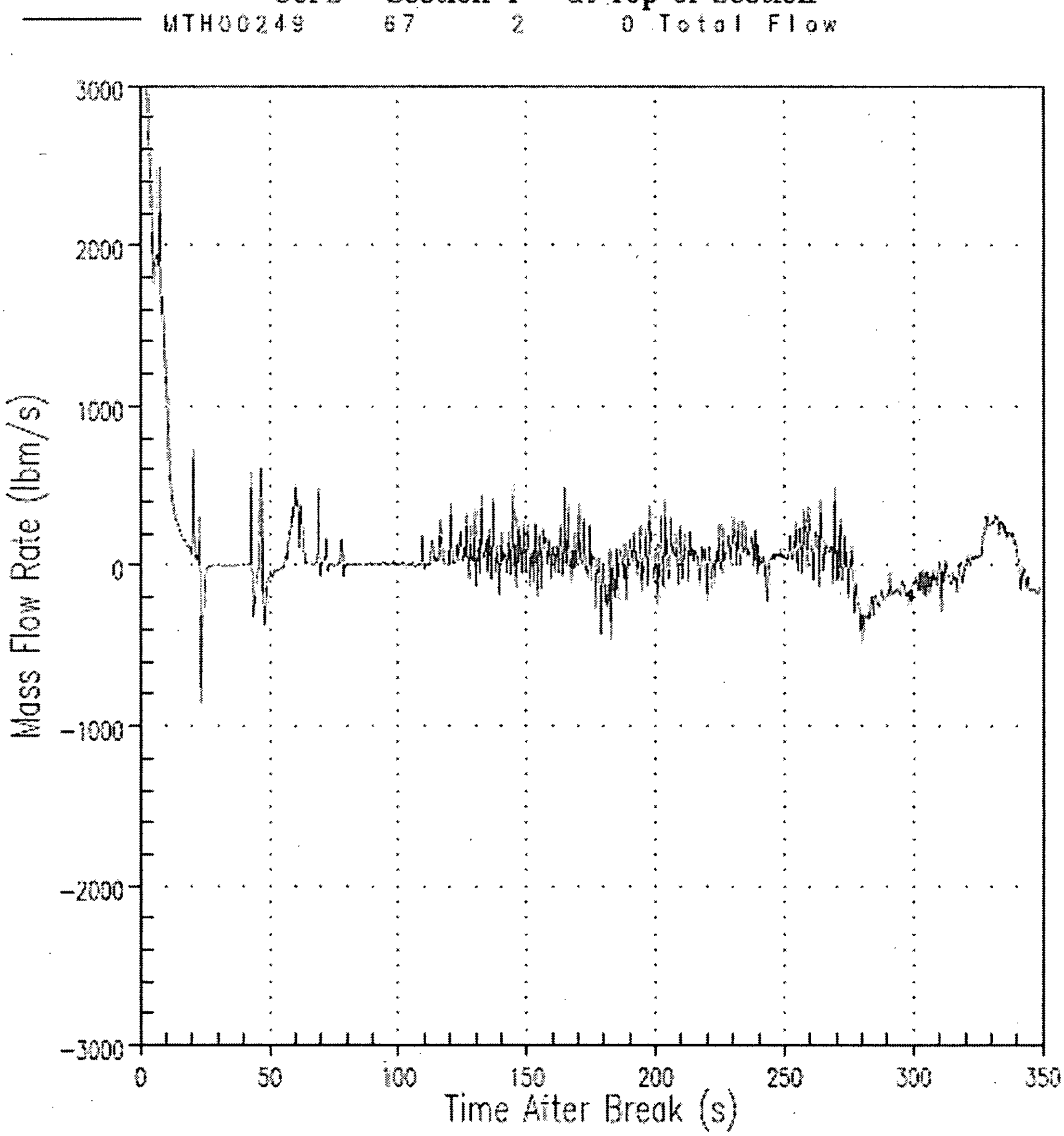
PTN Limiting PCT Case

Figure c43: Total Flow from Broken Loop Downcomer Channel 16
CCFL – Section 4 – at Top of Section



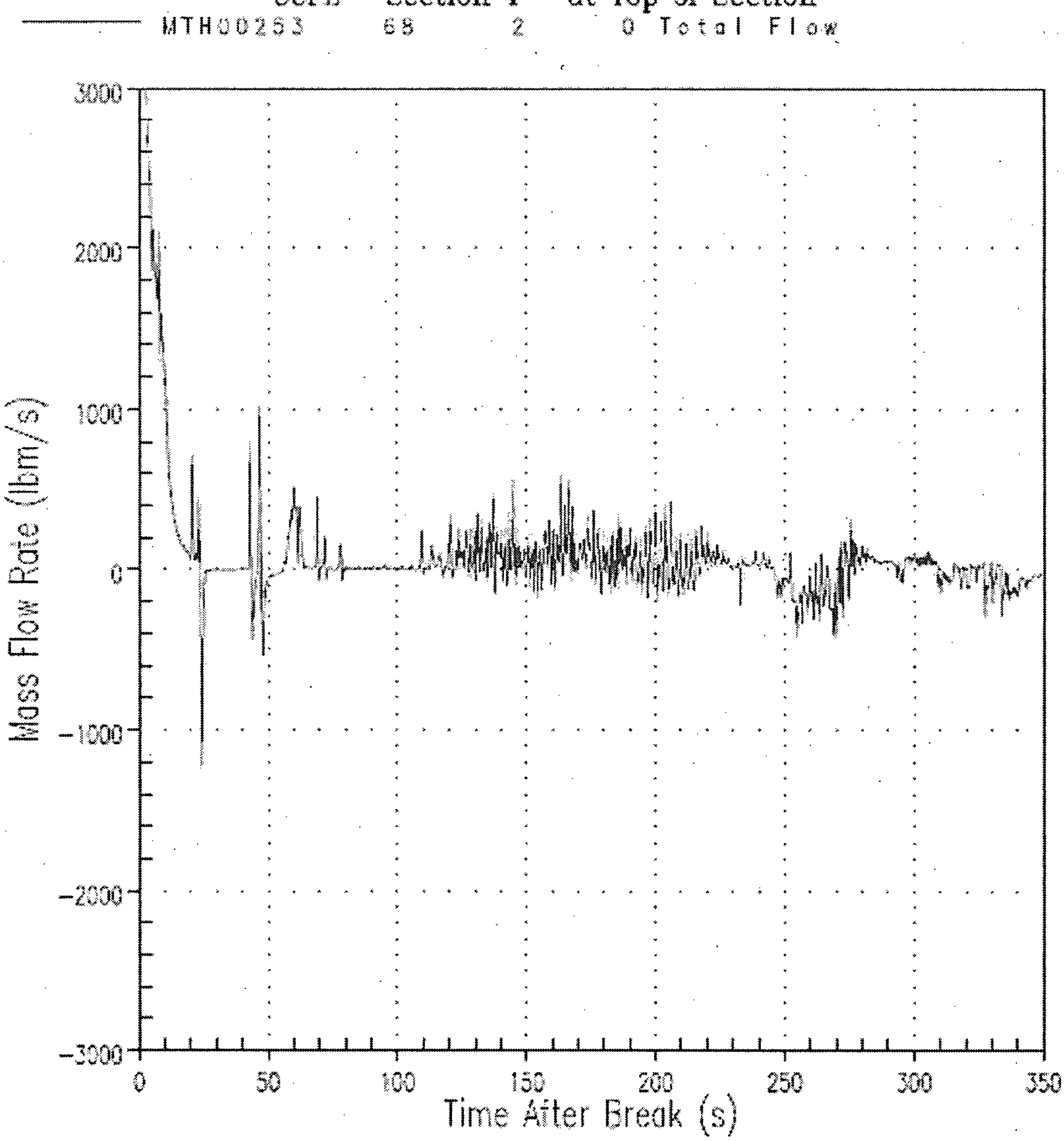
PTN Limiting PCT Case

Figure c44: Total Flow from Adjacen to Broken Loop Downcomer Channel 67
CCFL – Section 4 – at Top of Section



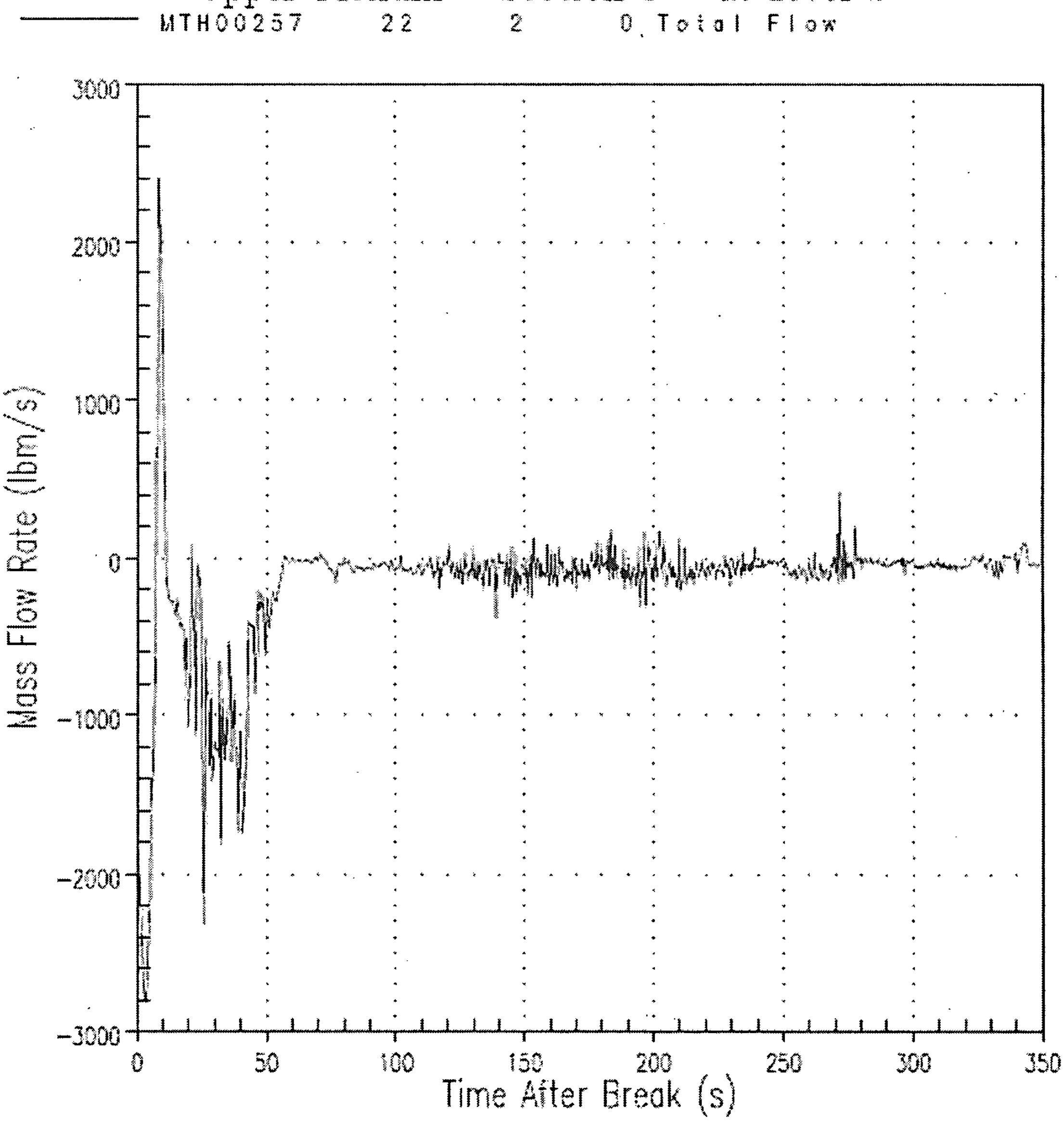
PTN Limiting PCT Case

Figure c45: Total Flow from Adjacen to Broken Loop Downcomer Channel 68
CCFL - Section 4 - at Top of Section



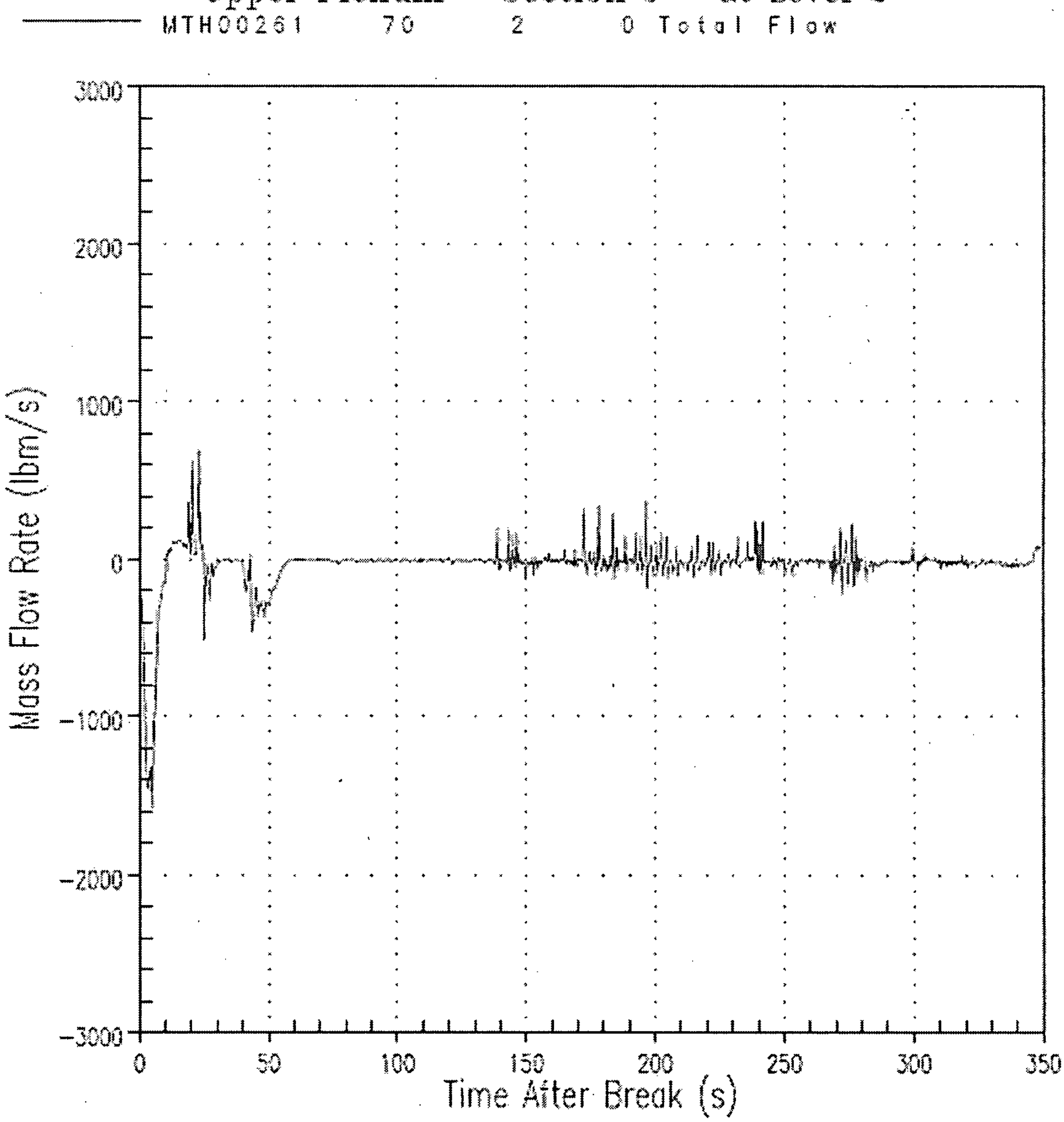
PTN Limiting PCT Case

Figure c46: Total Flow from Intact Loop Downcomer Channel 22 Upper Plenum – Section 5 – at Level 2



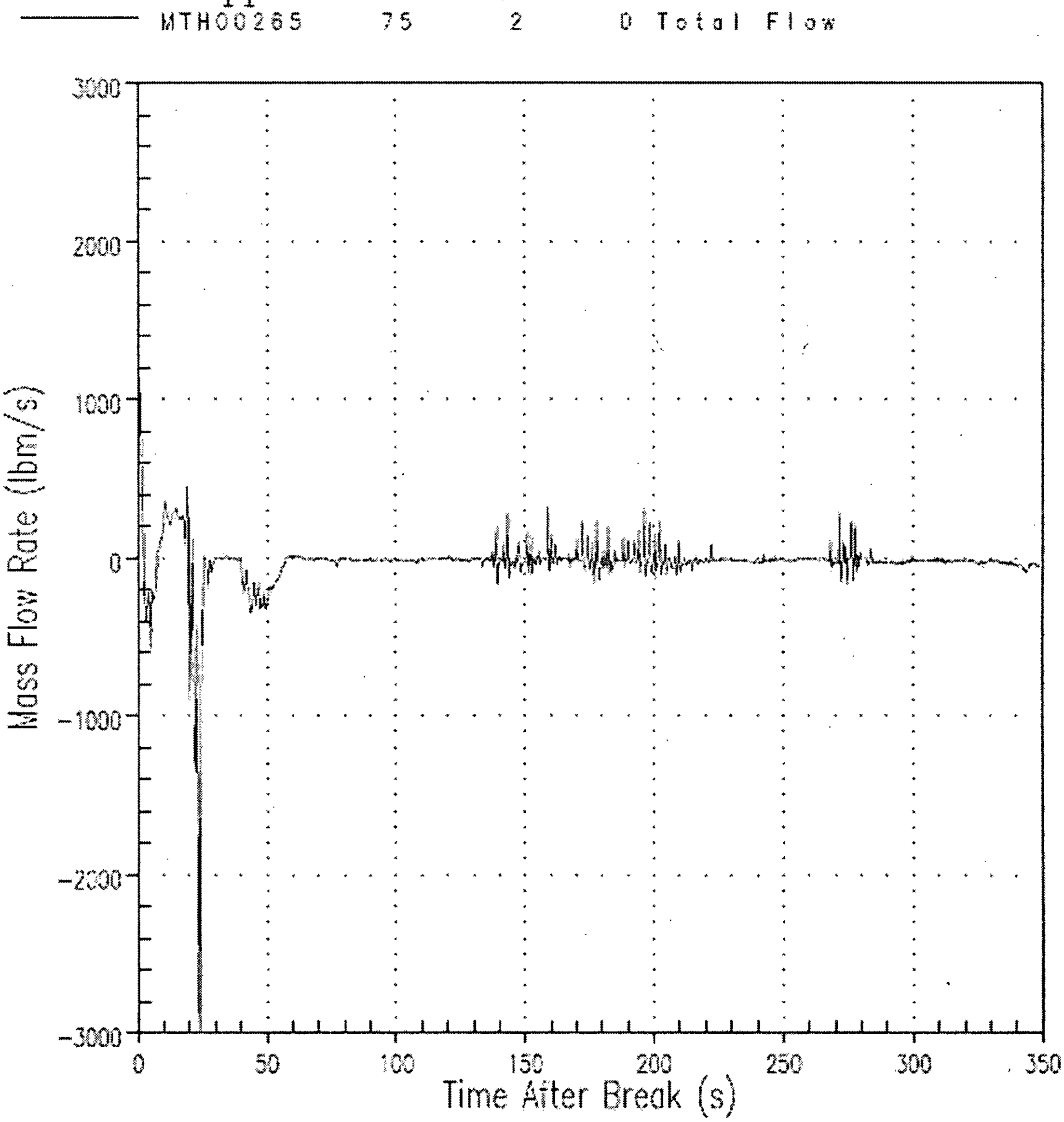
PTN Limiting PCT Case

Figure c47: Total Flow from Intact Loop Downcomer Channel 70 Upper Plenum – Section 5 – at Level 2



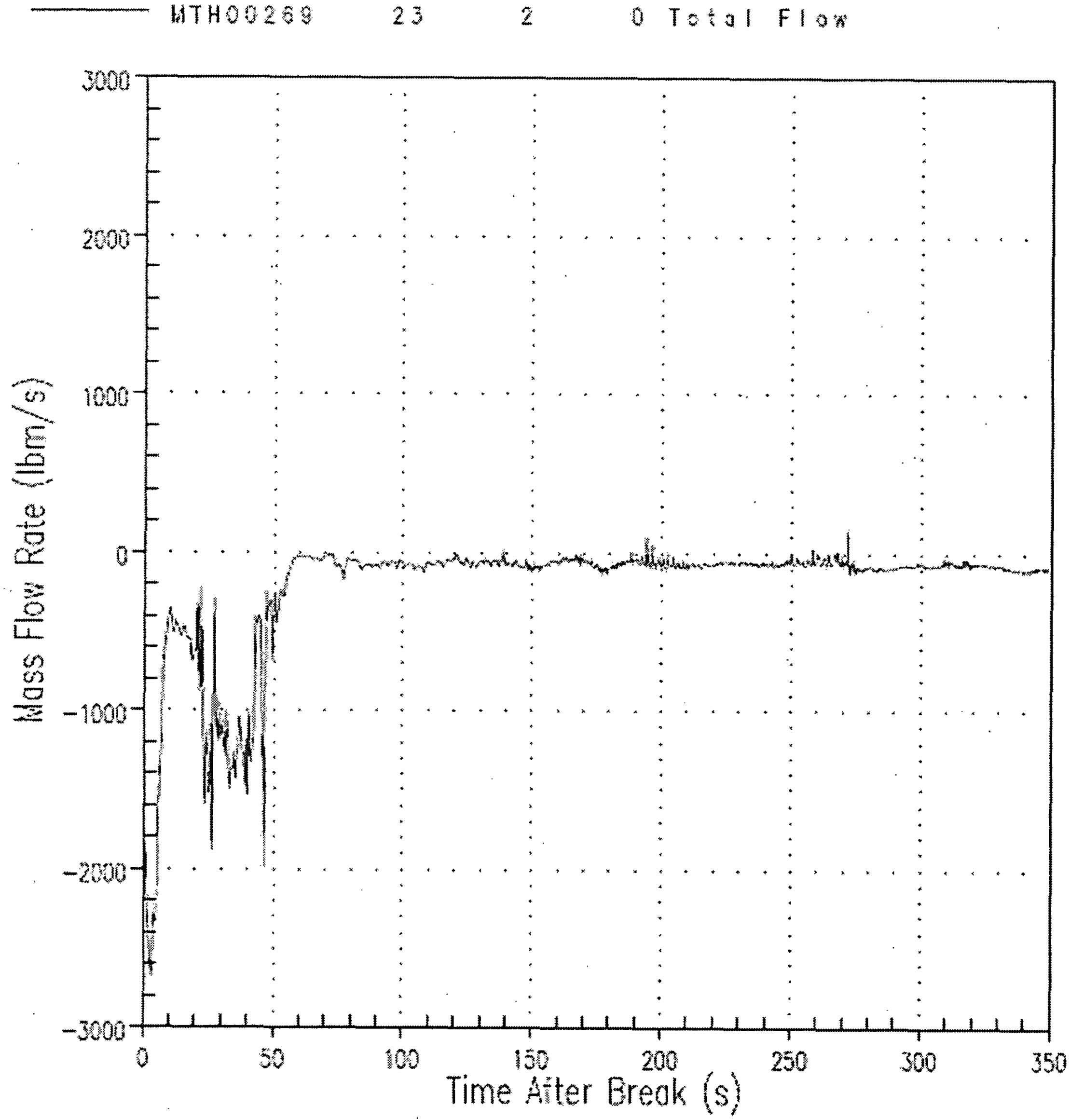
PTN Limiting PCT Case

Figure c48: Total Flow from Intact Loop Downcomer Channel 75
Upper Plenum – Section 5 – at Level 2



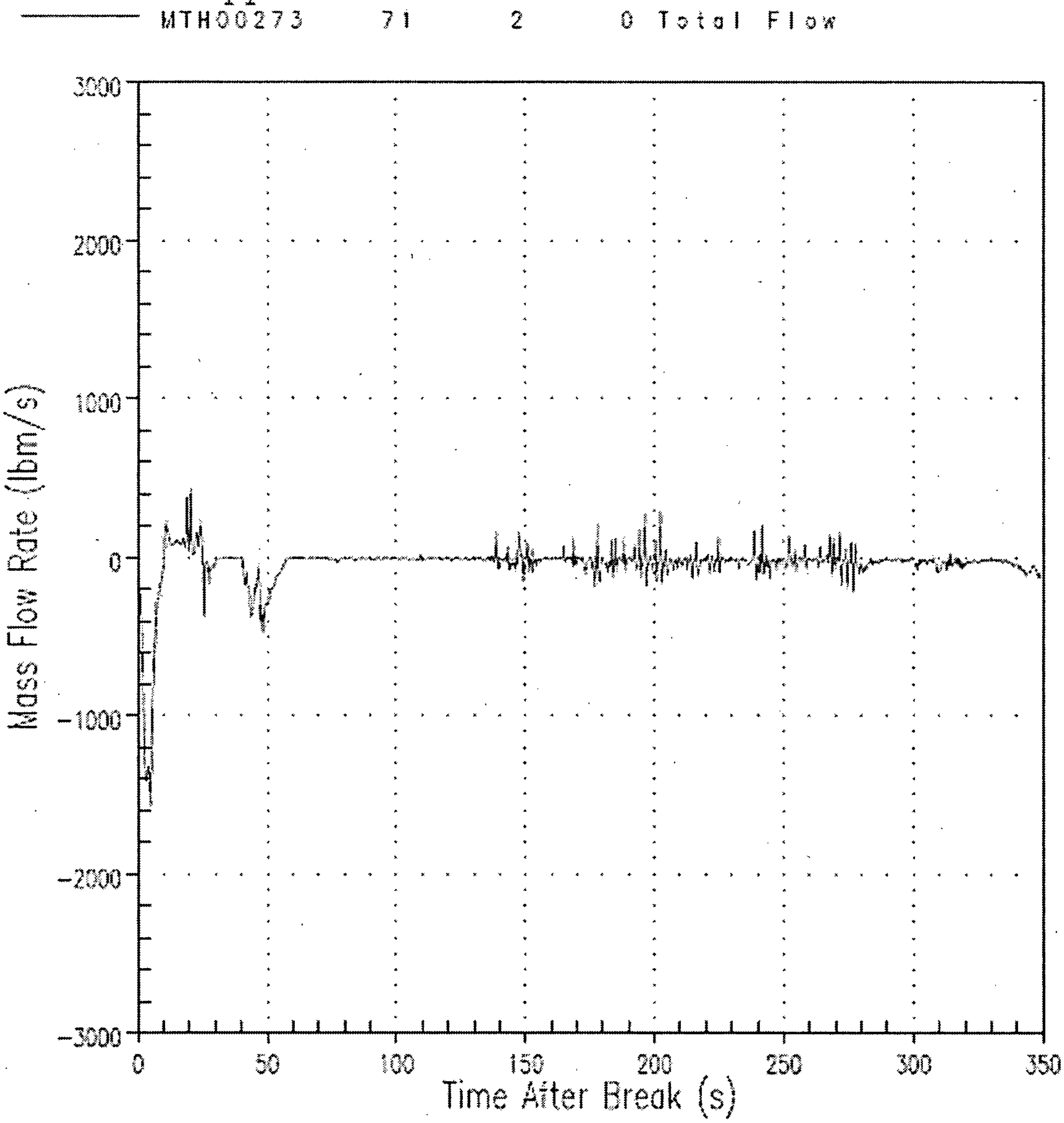
PTN Limiting PCT Case

Figure c49: Total Flow from Intact Loop Downcomer Channel 23 Upper Plenum - Section 5 - at Level 2



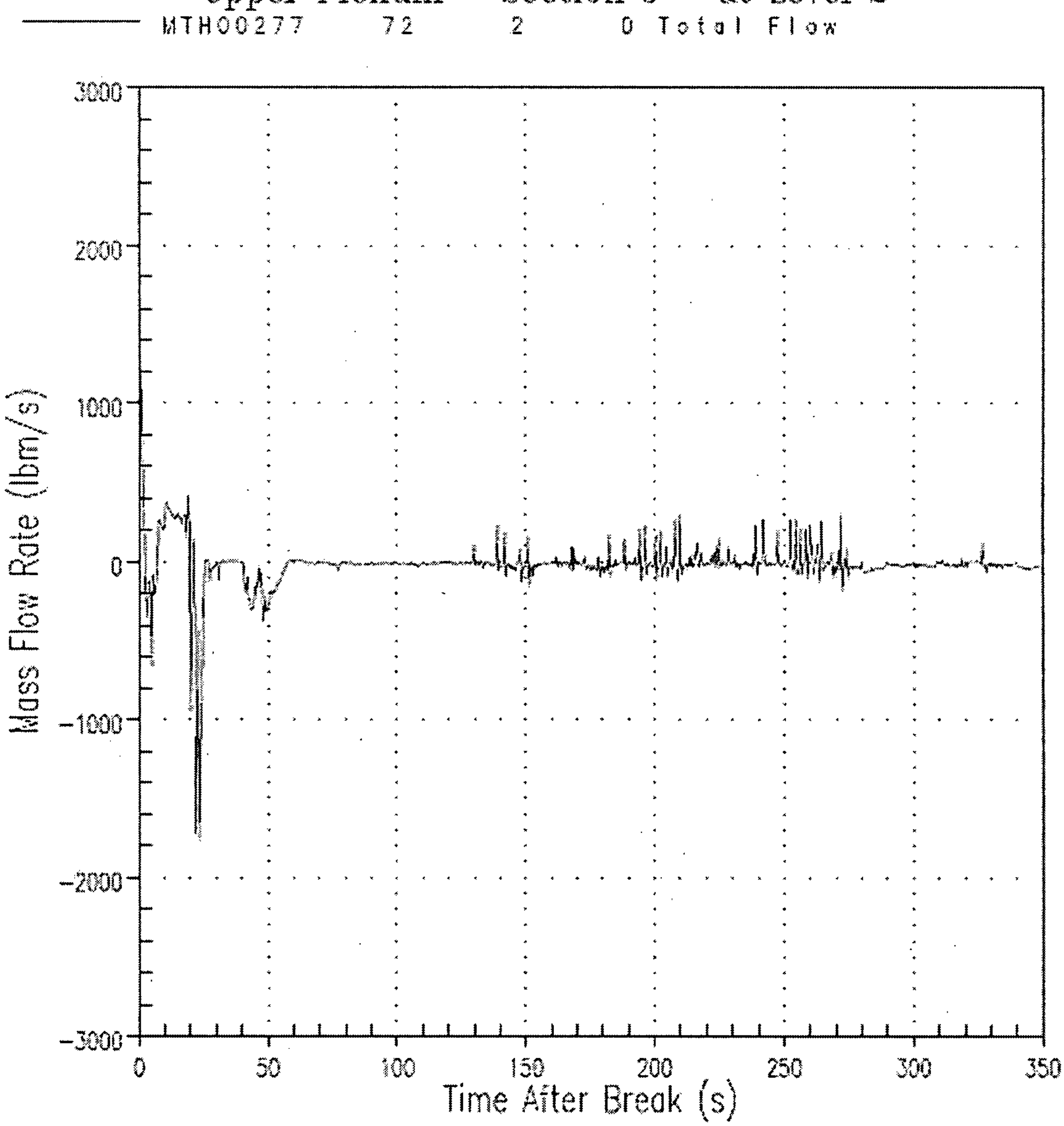
PTN Limiting PCT Case

Figure c50: Total Flow from Intact Loop Downcomer Channel 71 Upper Plenum - Section 5 - at Level 2



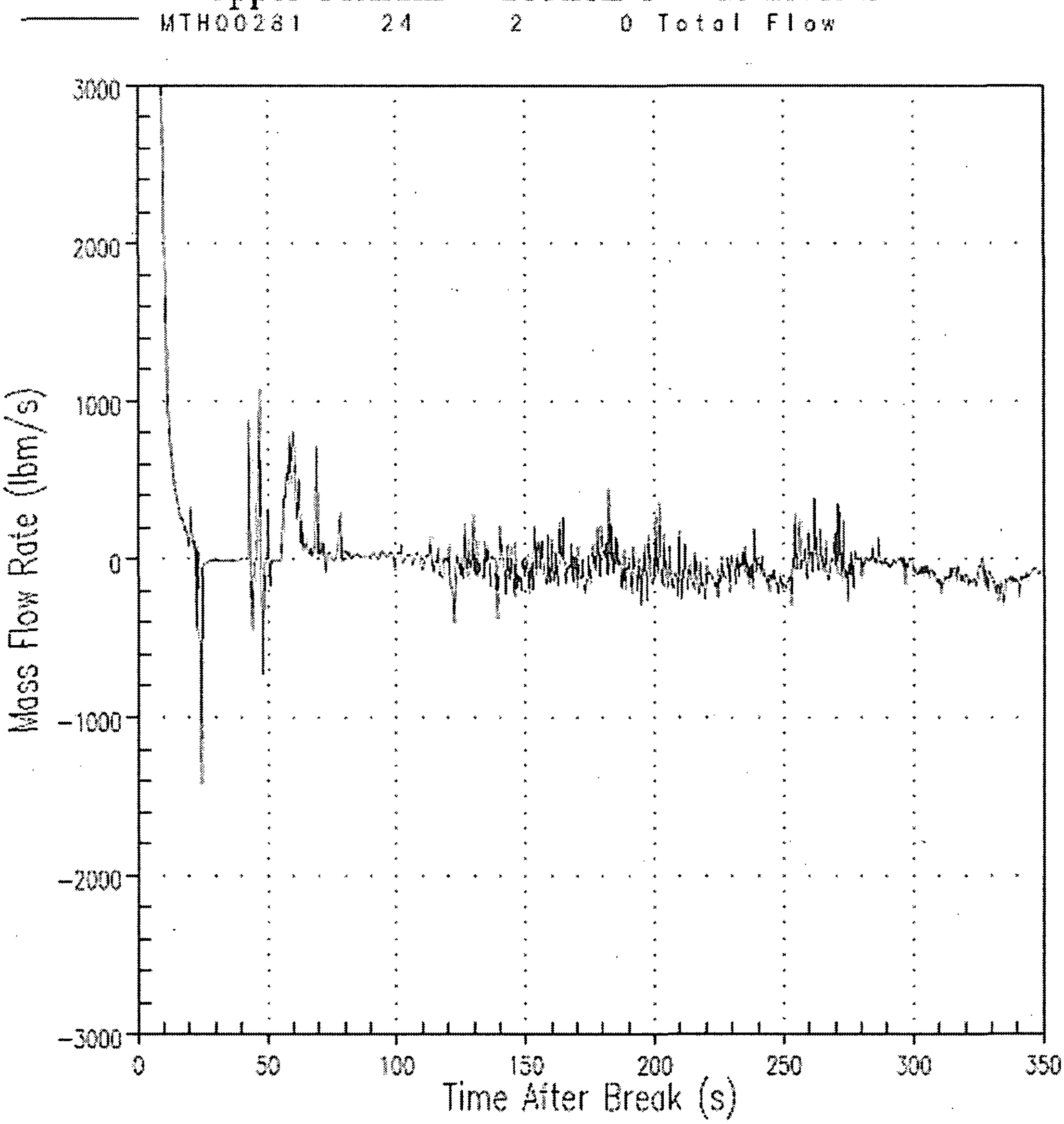
PTN Limiting PCT Case

Figure c51: Total Flow from Intact Loop Downcomer Channel 72
Upper Plenum – Section 5 – at Level 2



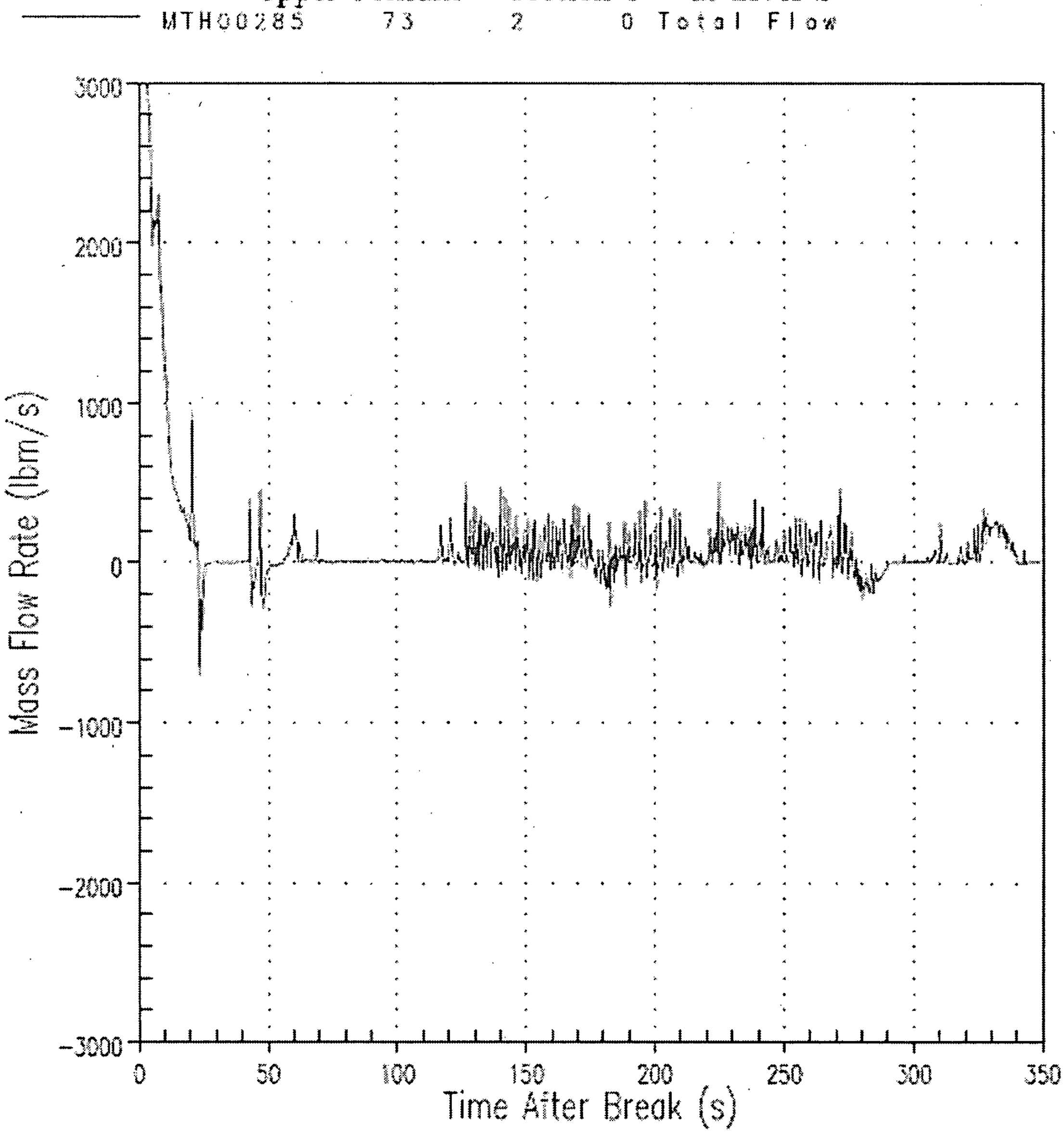
PTN Limiting PCT Case

Figure c52: Total Flow from Broken Loop Downcomer Channel 24 Upper Plenum – Section 5 – at Level 2



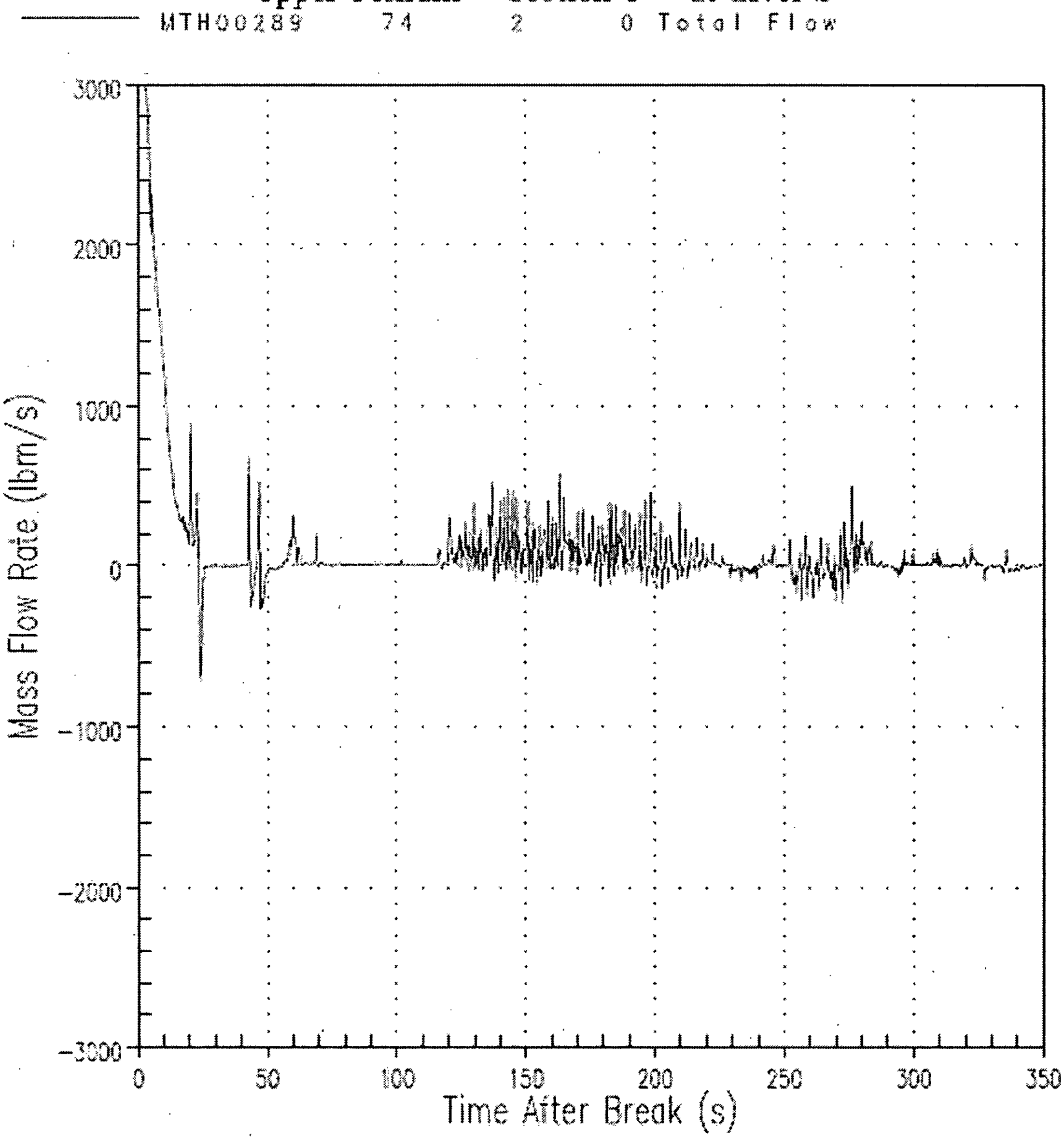
PTN Limiting PCT Case

Figure c53: Total Flow from Adjacen to Broken Loop Downcomer Channel 73
Upper Plenum – Section 5 – at Level 2



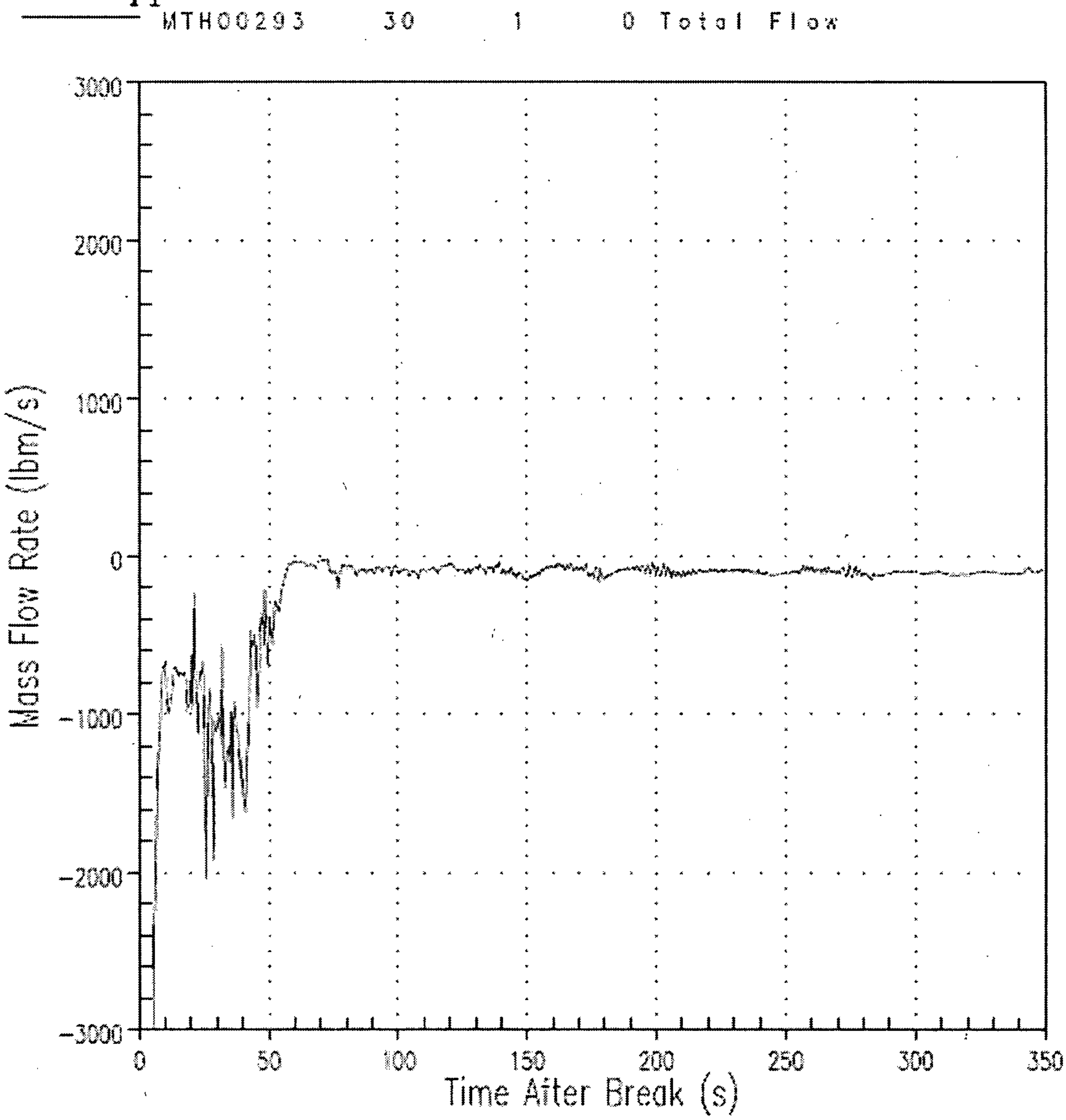
PTN Limiting PCT Case

Figure c54: Total Flow from Adjacen to Broken Loop Downcomer Channel 74 Upper Plenum – Section 5 – at Level 2



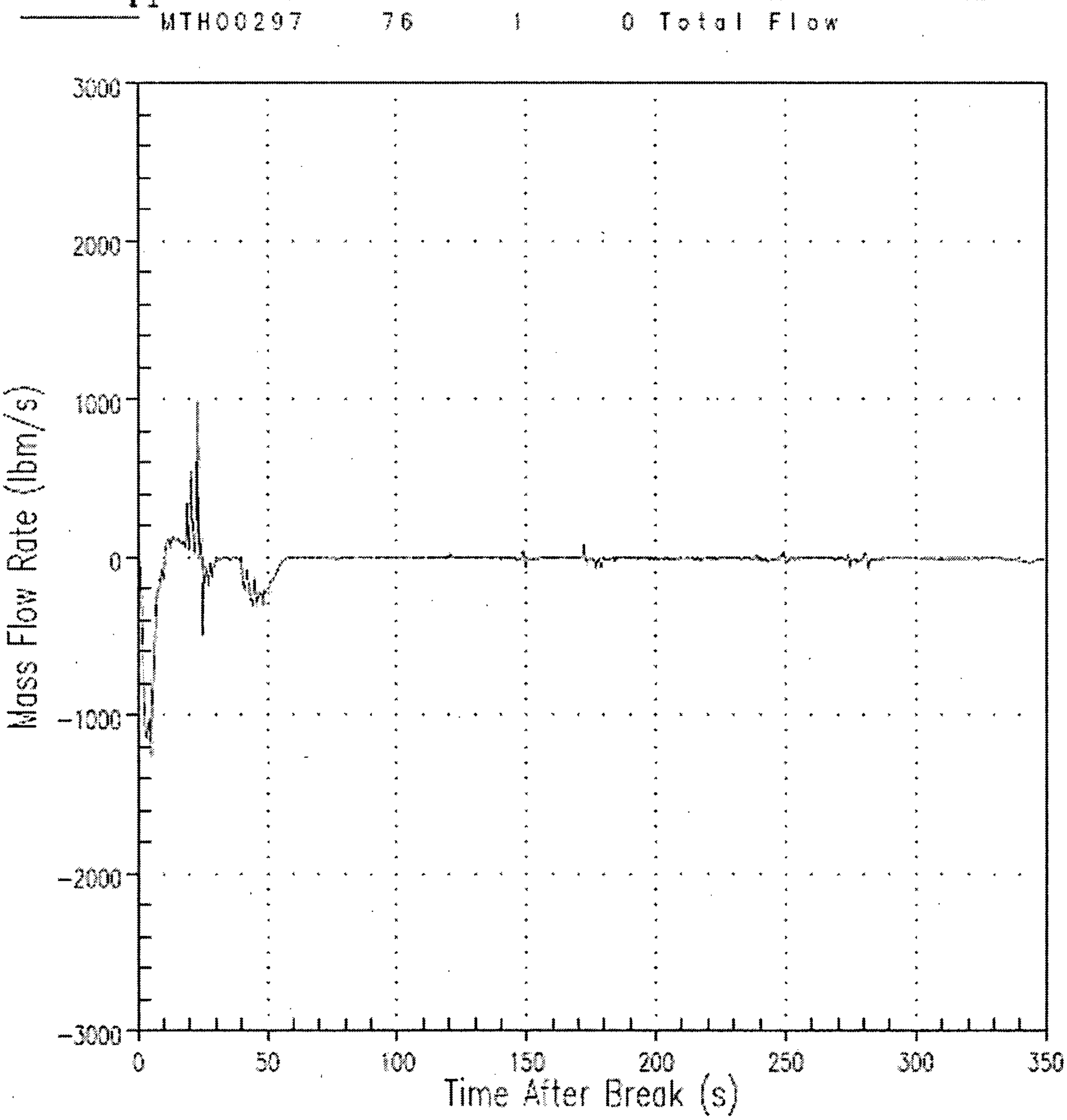
PTN Limiting PCT Case

Figure c55: Total Flow from Intact Loop Downcomer Channel 30
Upper Plenum - Section 6 - at Bottom of Section



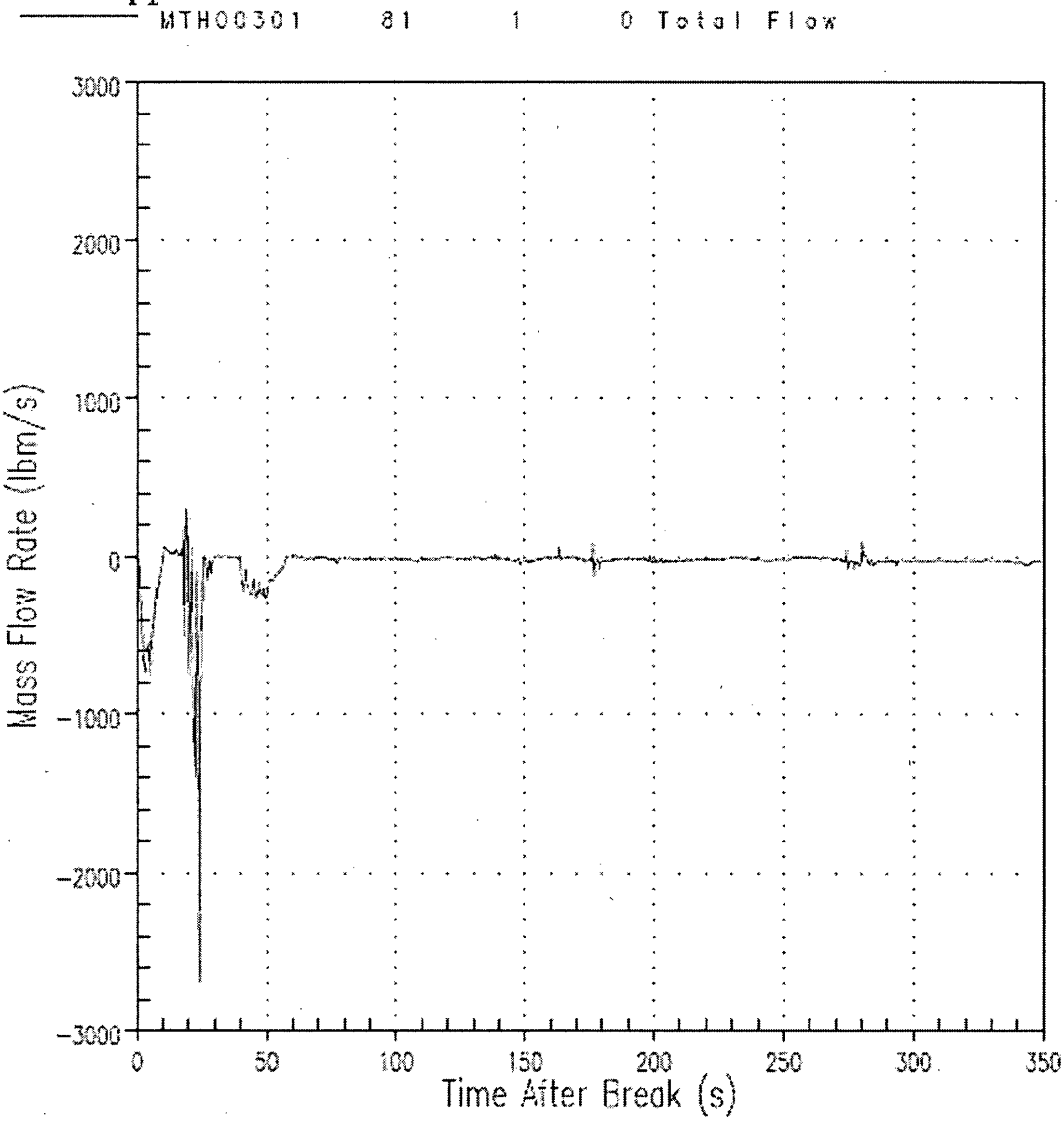
PTN Limiting PCT Case

Figure c56: Total Flow from Intact Loop Downcomer Channel 76
Upper Plenum – Section 6 – at Bottom of Section



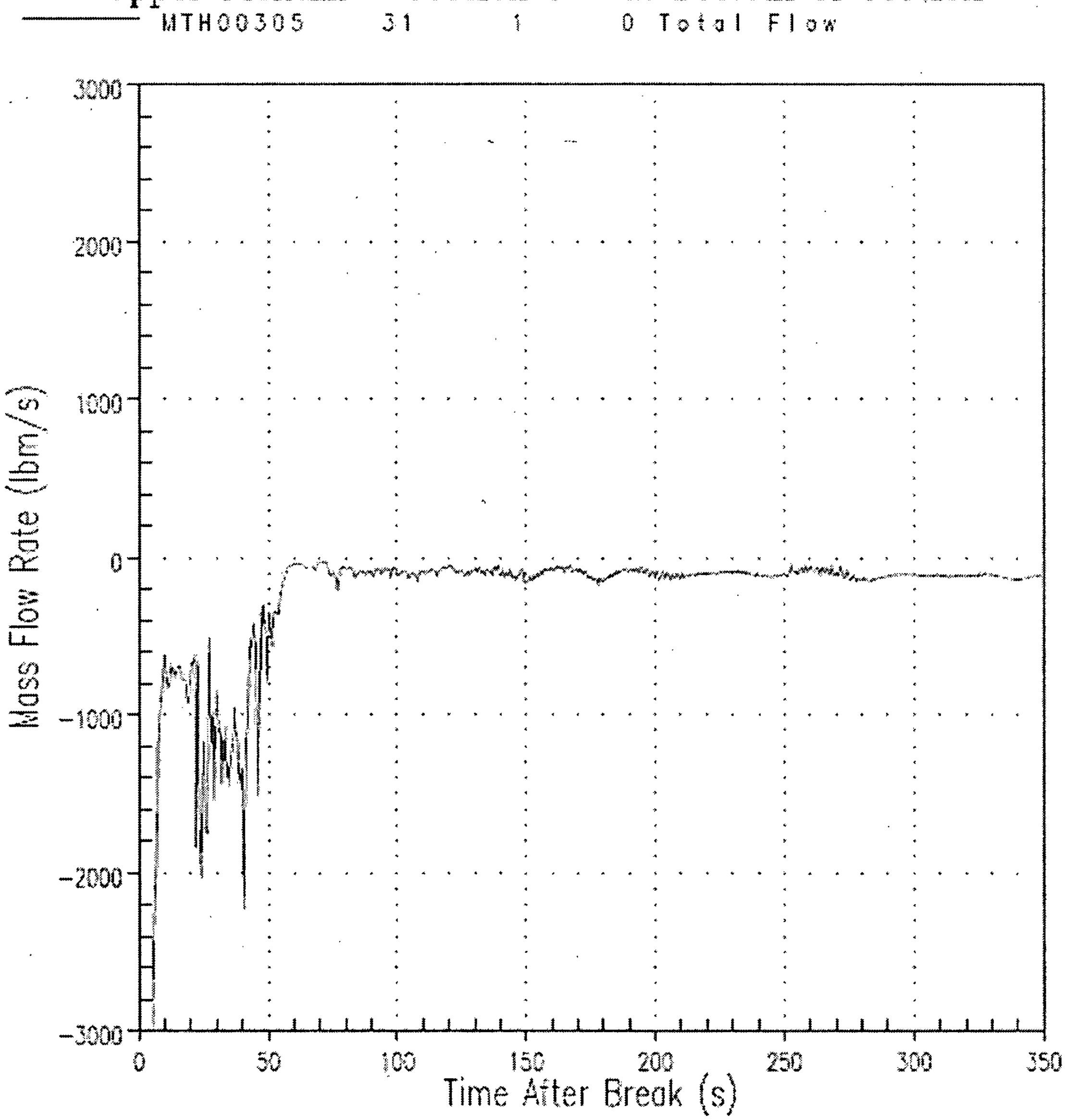
PTN Limiting PCT Case

Figure c57: Total Flow from Intact Loop Downcomer Channel 81 Upper Plenum – Section 6 – at Bottom of Section

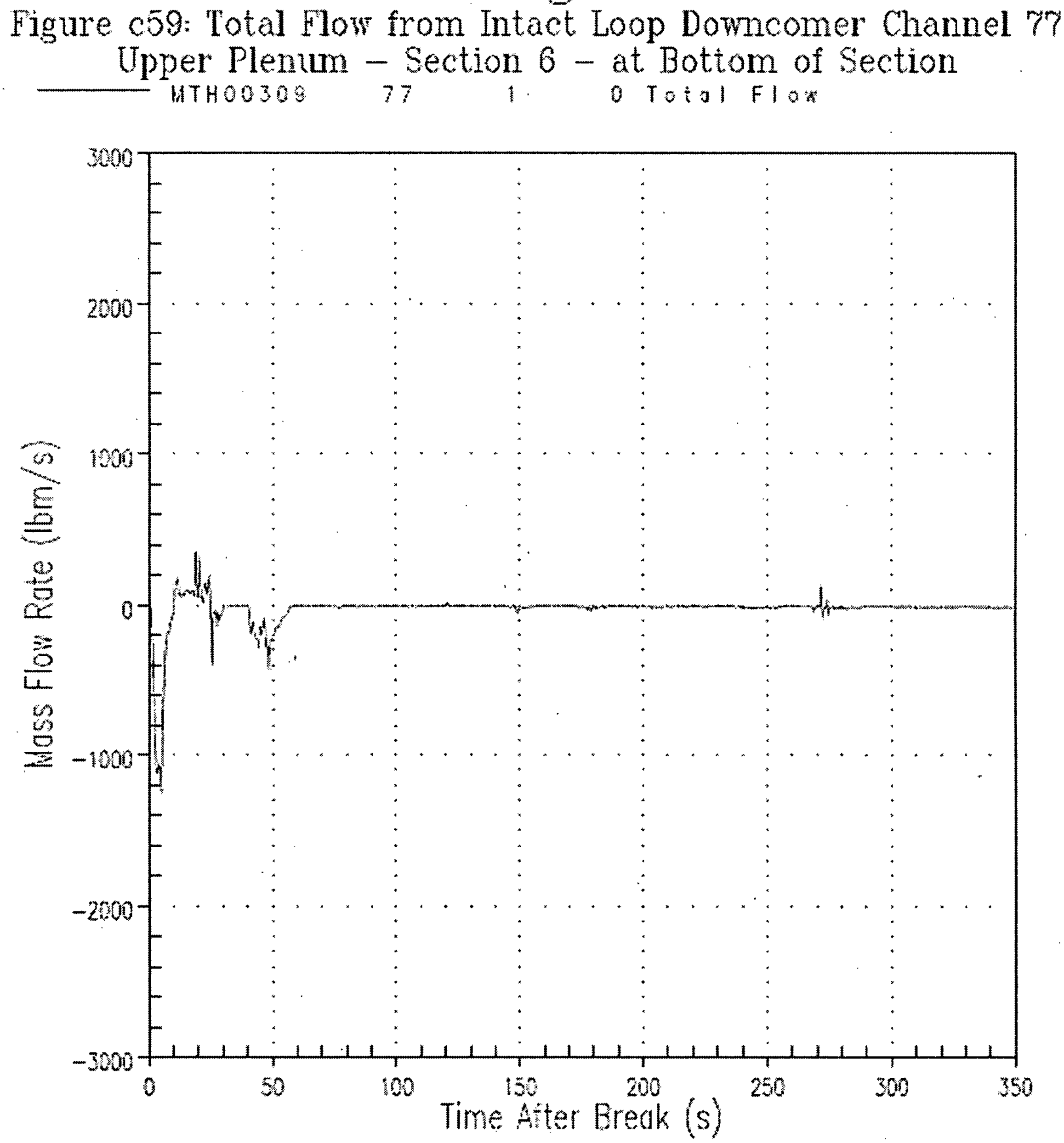


PTN Limiting PCT Case

Figure c58: Total Flow from Intact Loop Downcomer Channel 31
Upper Plenum – Section 6 – at Bottom of Section

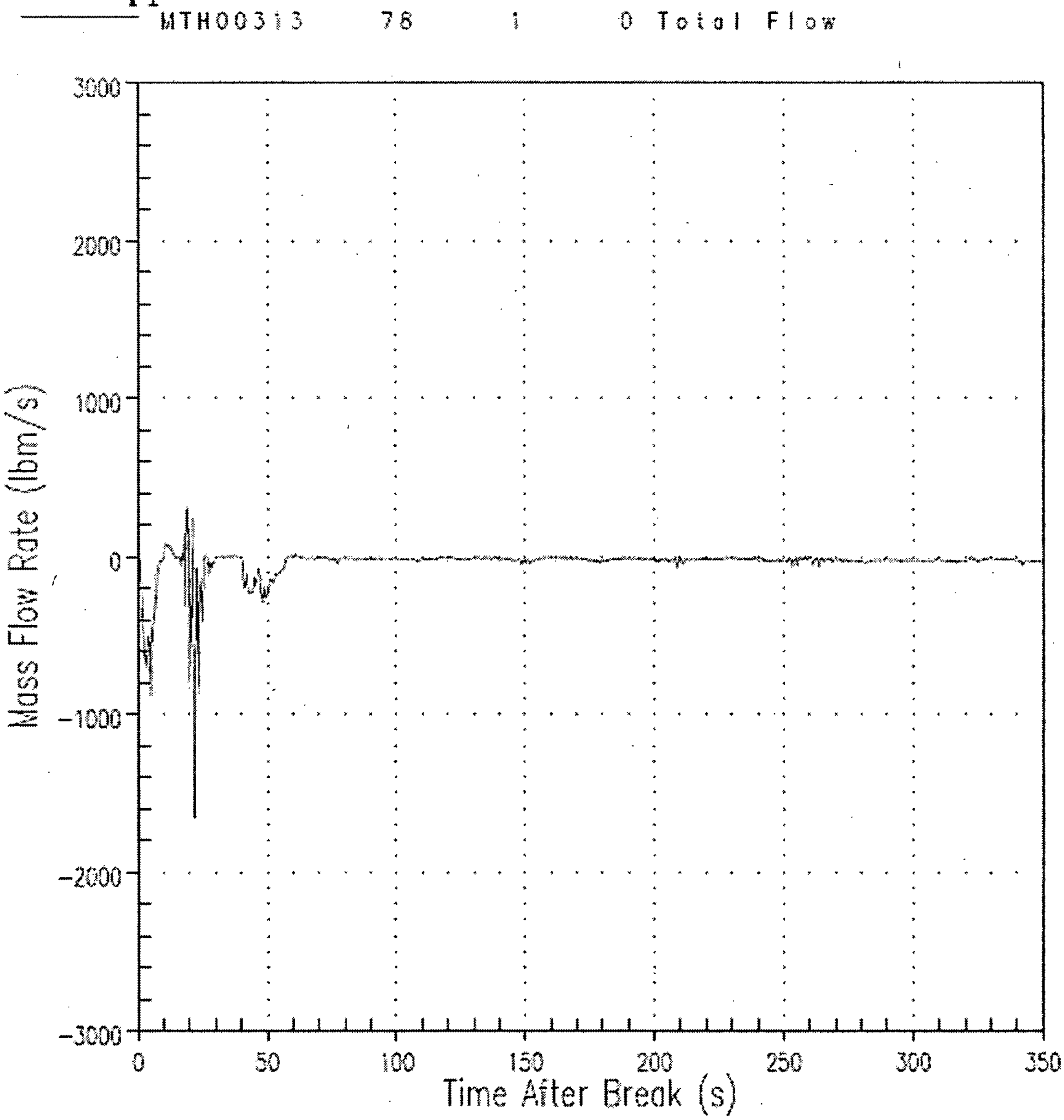


PTN Limiting PCT Case



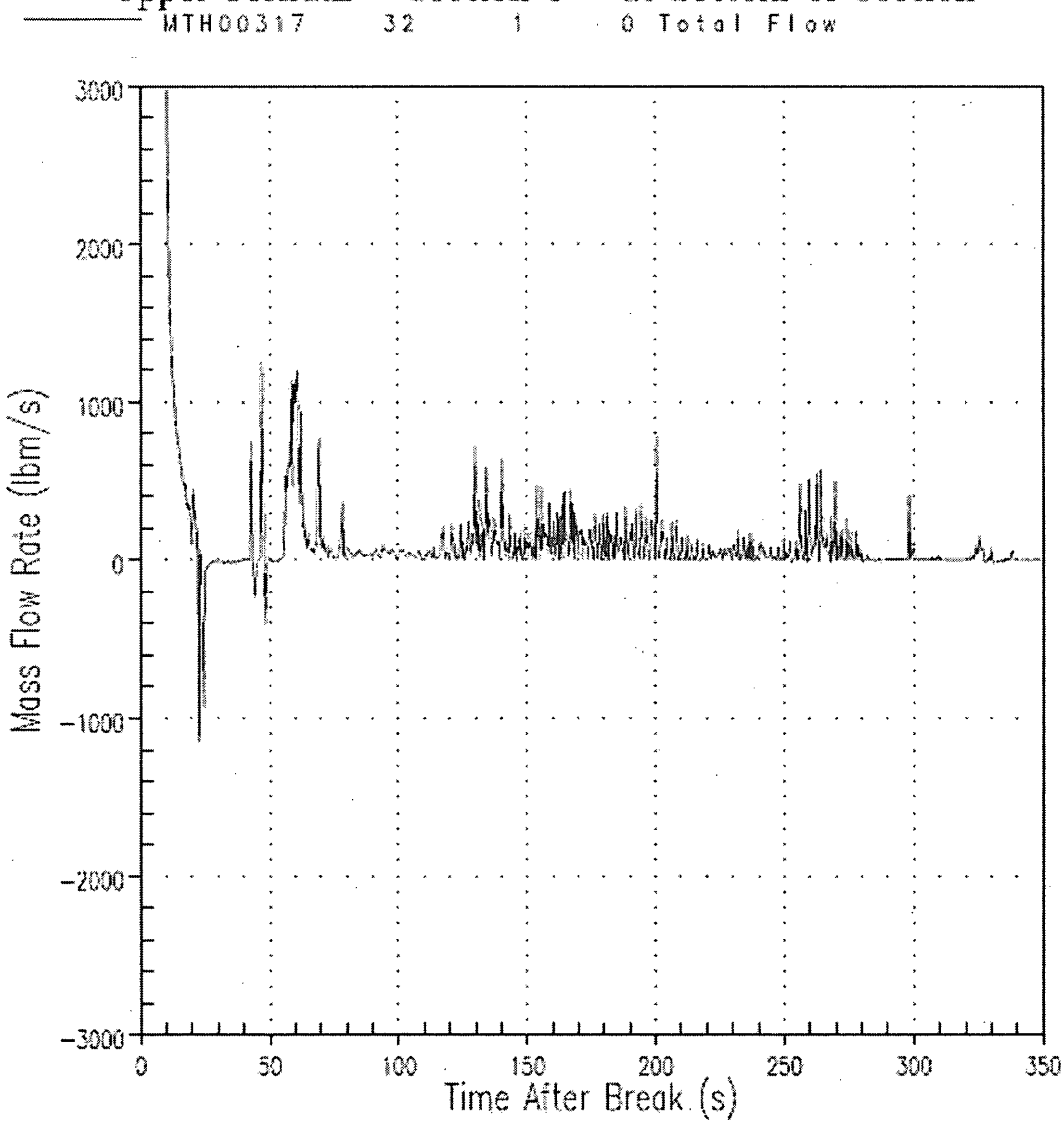
PTN Limiting PCT Case

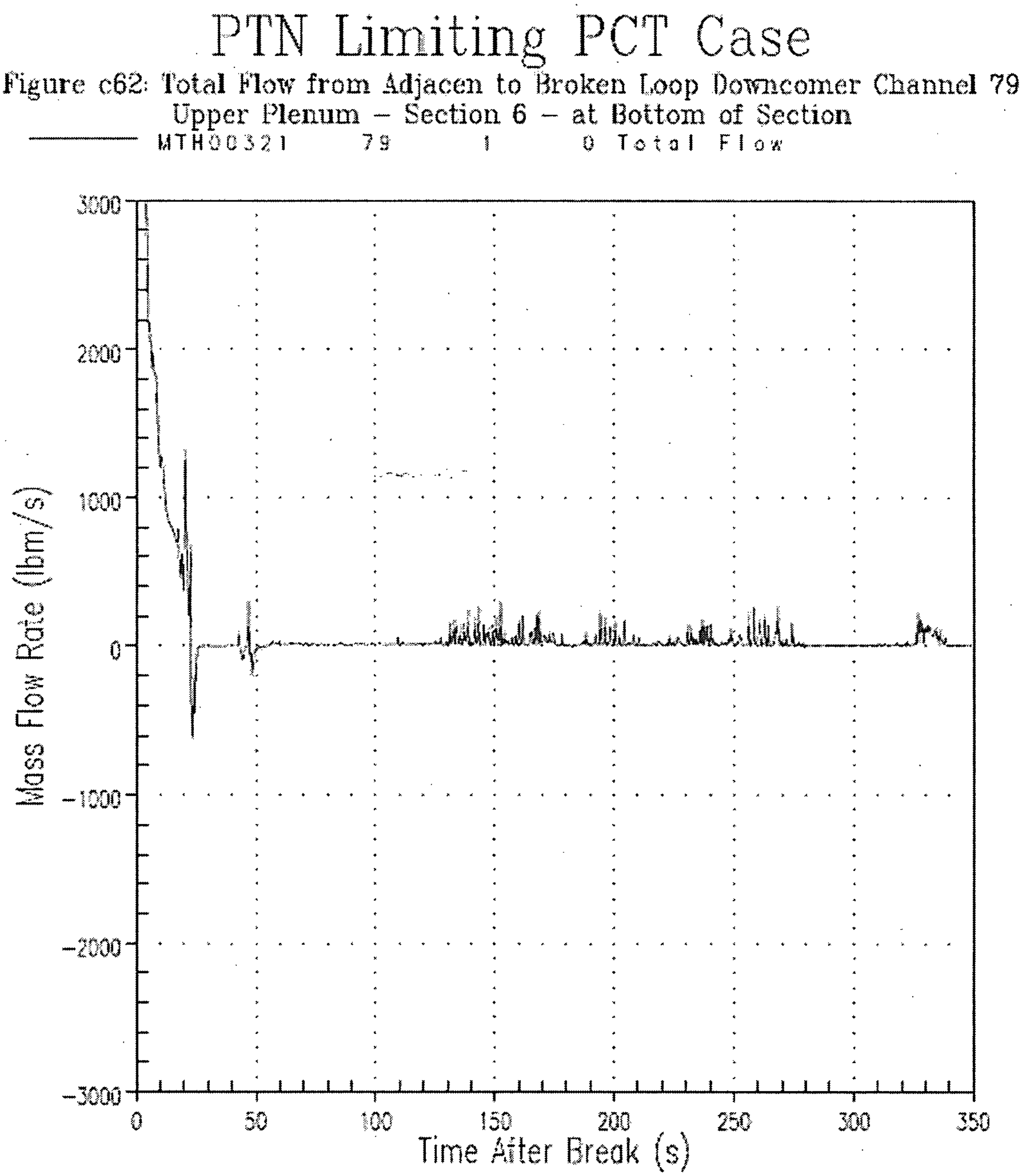
Figure c60: Total Flow from Intact Loop Downcomer Channel 78
Upper Plenum – Section 6 – at Bottom of Section



PTN Limiting PCT Case

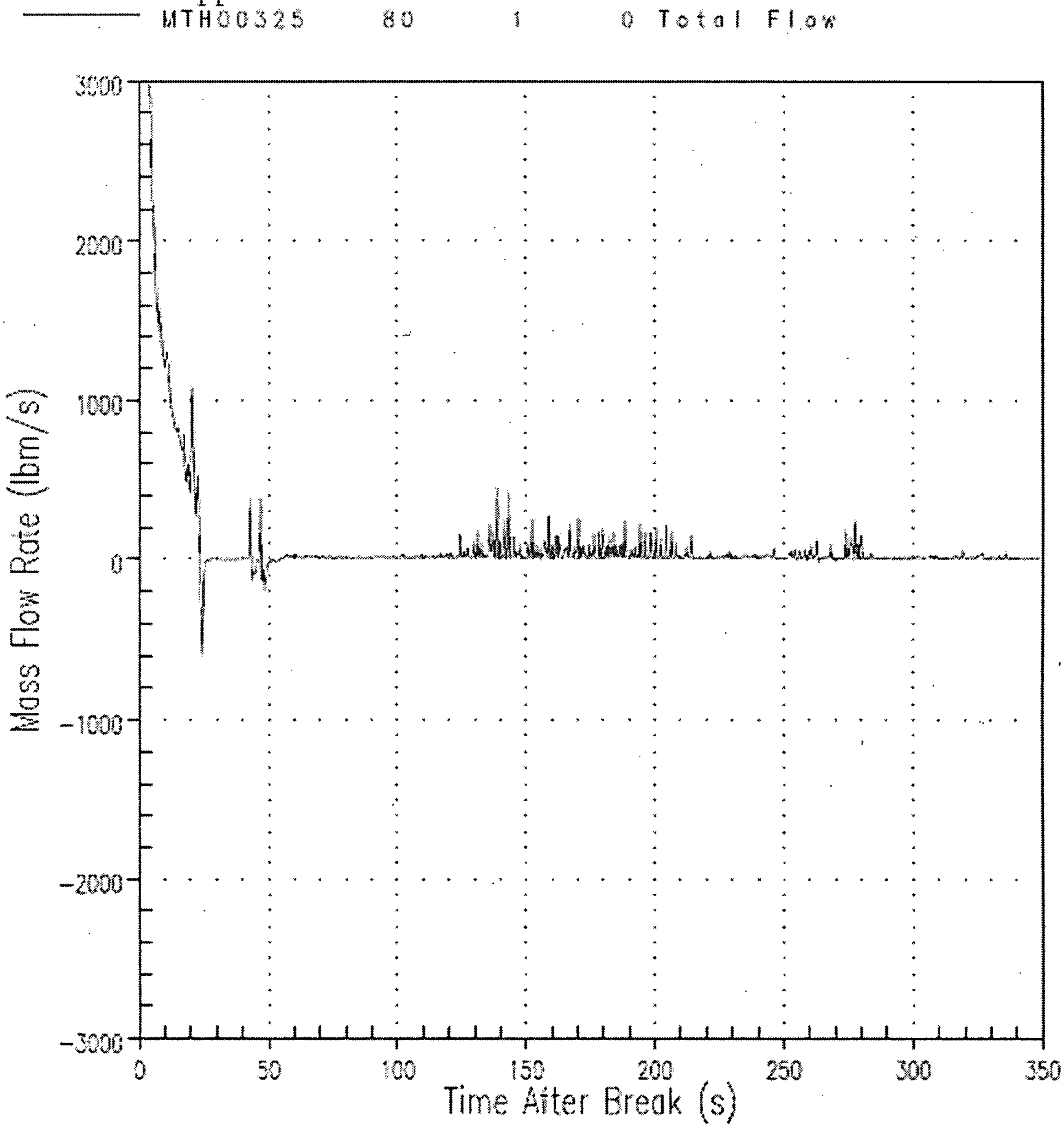
Figure c61: Total Flow from Broken Loop Downcomer Channel 32 Upper Plenum – Section 6 – at Bottom of Section




PTN Limiting PCT Case
Figure c62: Total Flow from Adjacen to Broken Loop Downcomer Channel 79
Upper Plenum – Section 6 – at Bottom of Section
MTH00321 79 1 0 Total Flow
3000
2000
1000
0
-1000
-2000
-3000
Mass Flow Rate (lbm/s)
0
50
100
150
200
250
300
350
Time After Break (s)

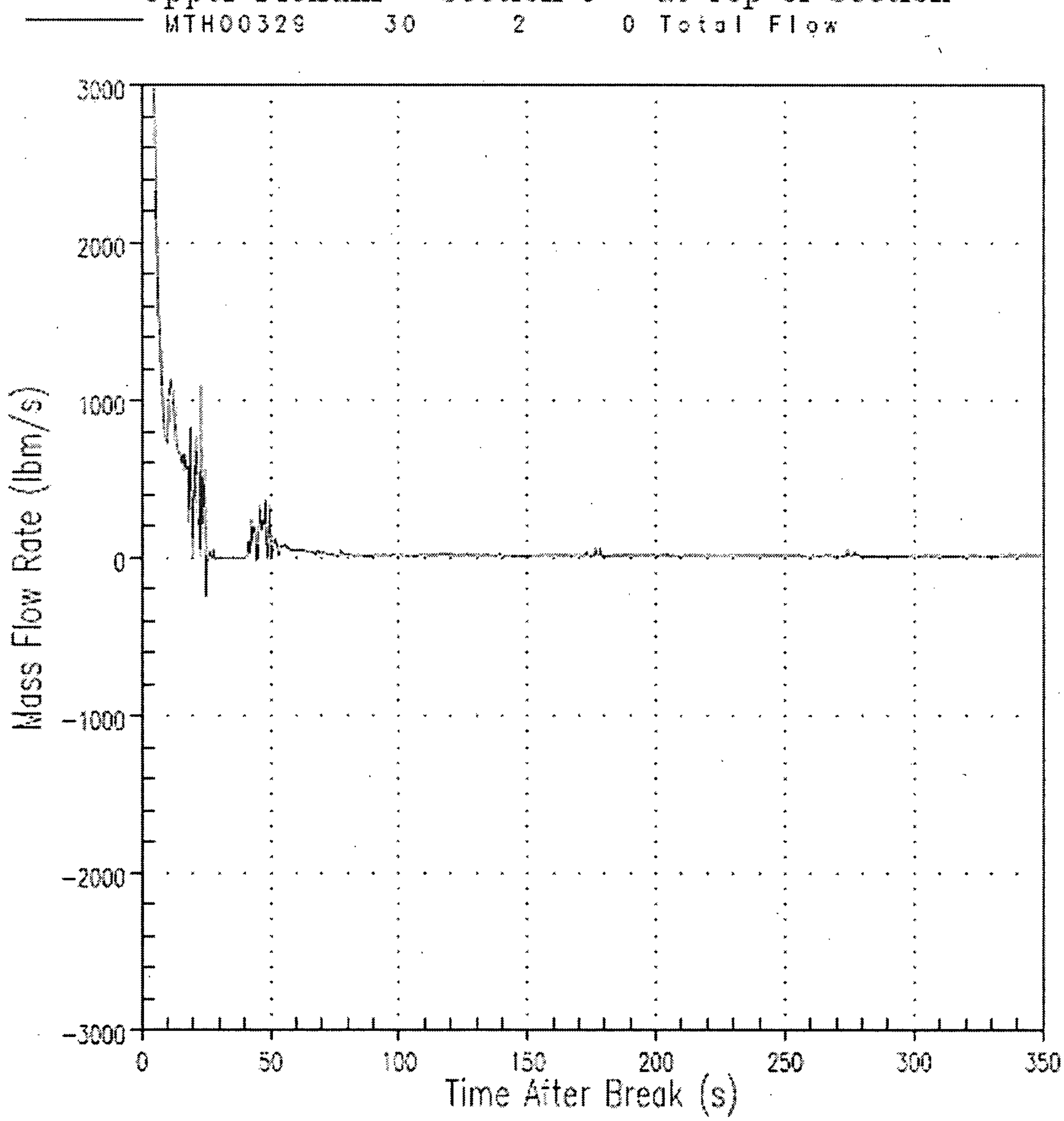
PTN Limiting PCT Case

Figure c63: Total Flow from Adjacen to Broken Loop Downcomer Channel 80
Upper Plenum – Section 6 – at Bottom of Section



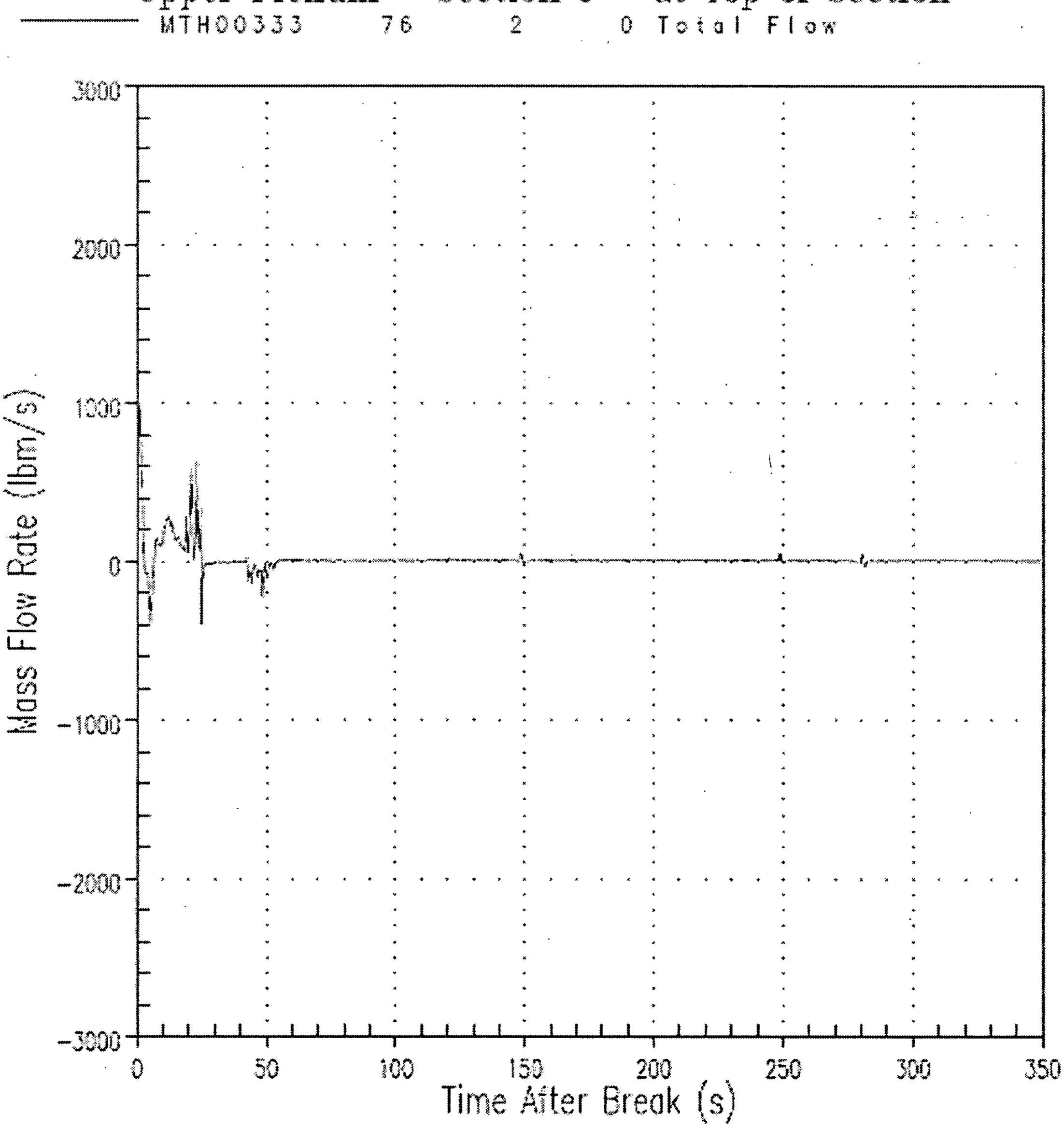
PTN Limiting PCT Case

Figure c64: Total Flow from Intact Loop Downcomer Channel 30 Upper Plenum – Section 6 – at Top of Section



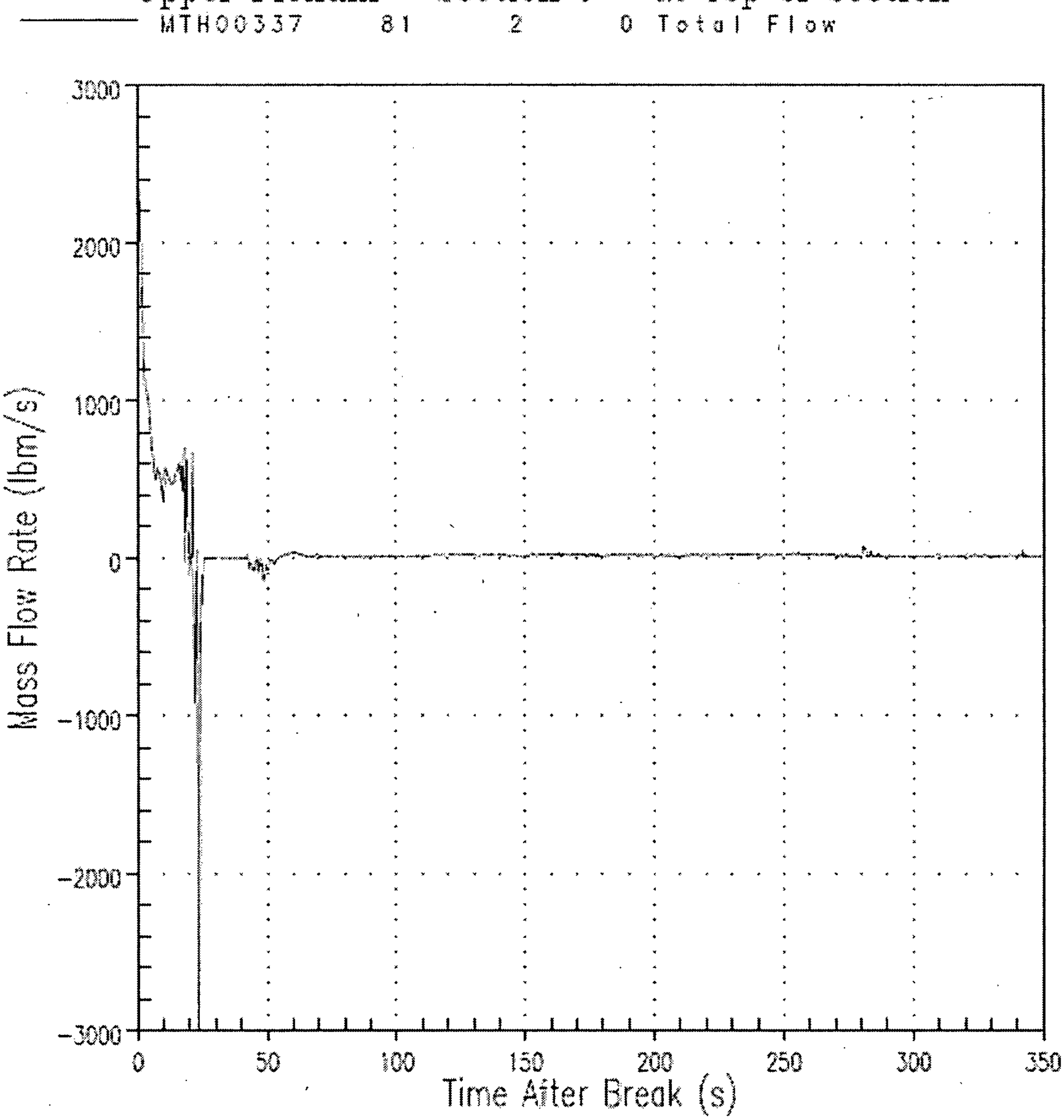
PTN Limiting PCT Case

Figure c65: Total Flow from Intact Loop Downcomer Channel 76
Upper Plenum – Section 6 – at Top of Section



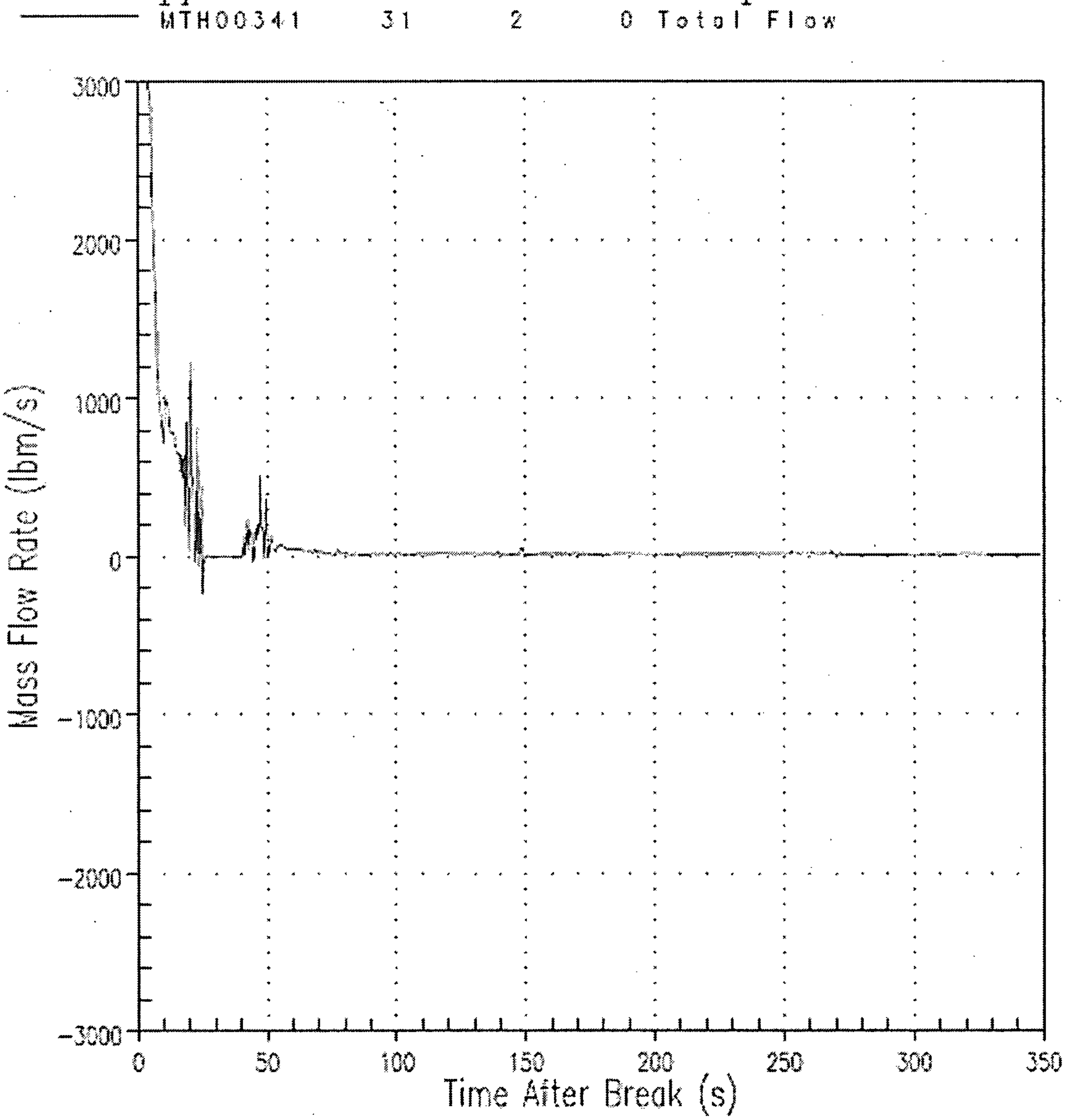
PTN Limiting PCT Case

Figure c66: Total Flow from Intact Loop Downcomer Channel 81
Upper Plenum – Section 6 – at Top of Section



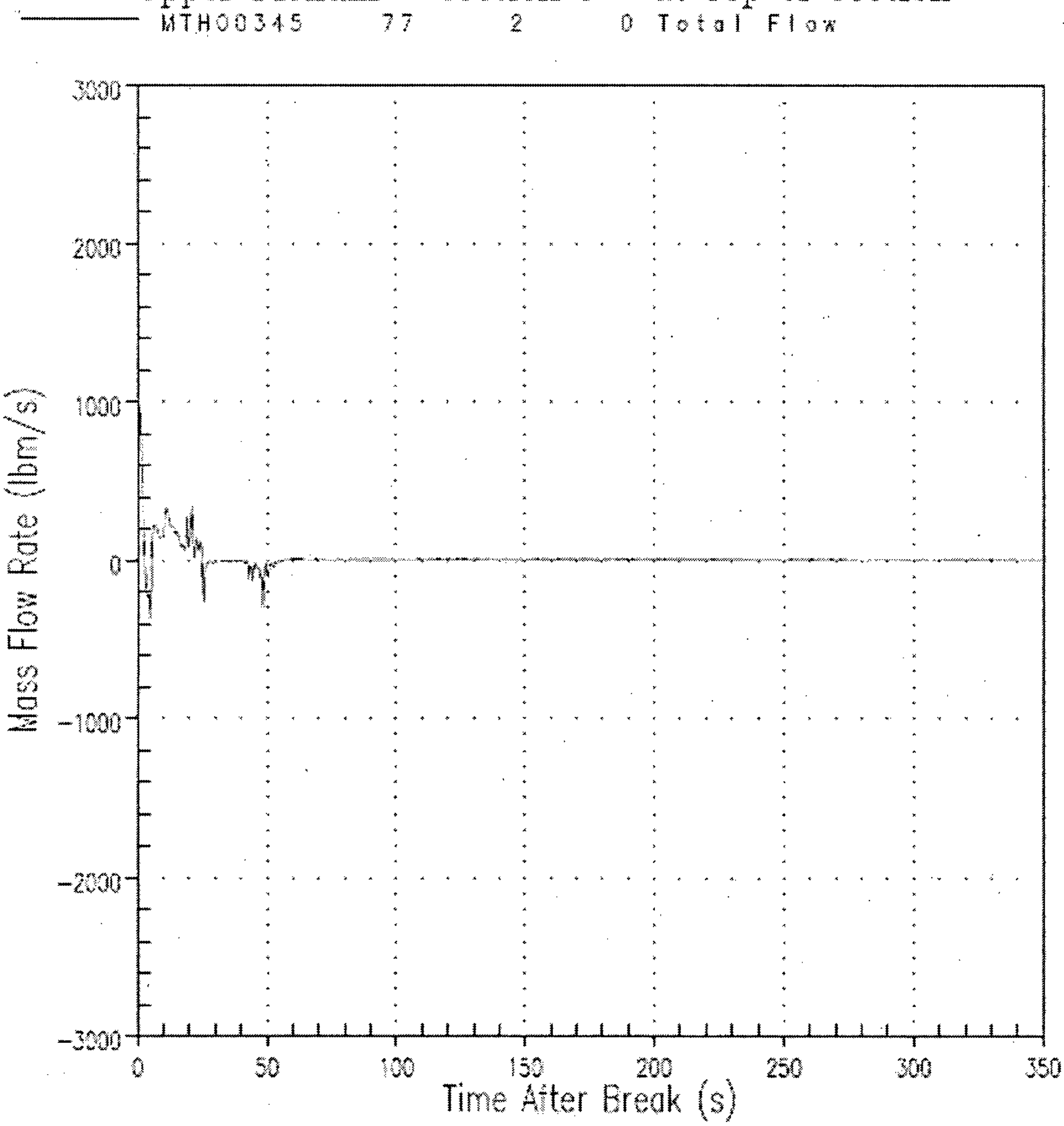
PTN Limiting PCT Case

Figure c67: Total Flow from Intact Loop Downcomer Channel 31
Upper Plenum – Section 6 – at Top of Section



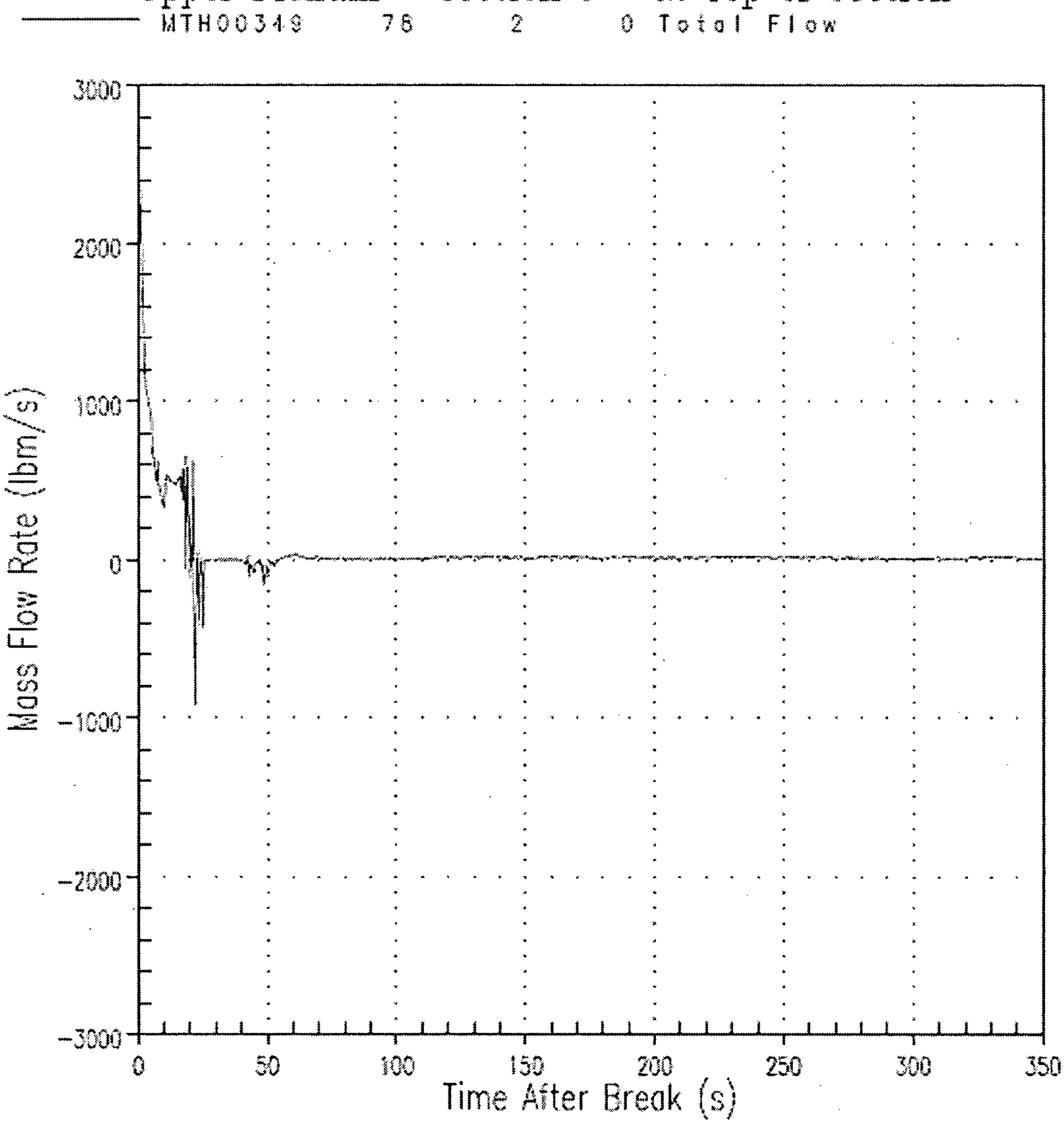
PTN Limiting PCT Case

Figure c68: Total Flow from Intact Loop Downcomer Channel 77
Upper Plenum - Section 6 - at Top of Section



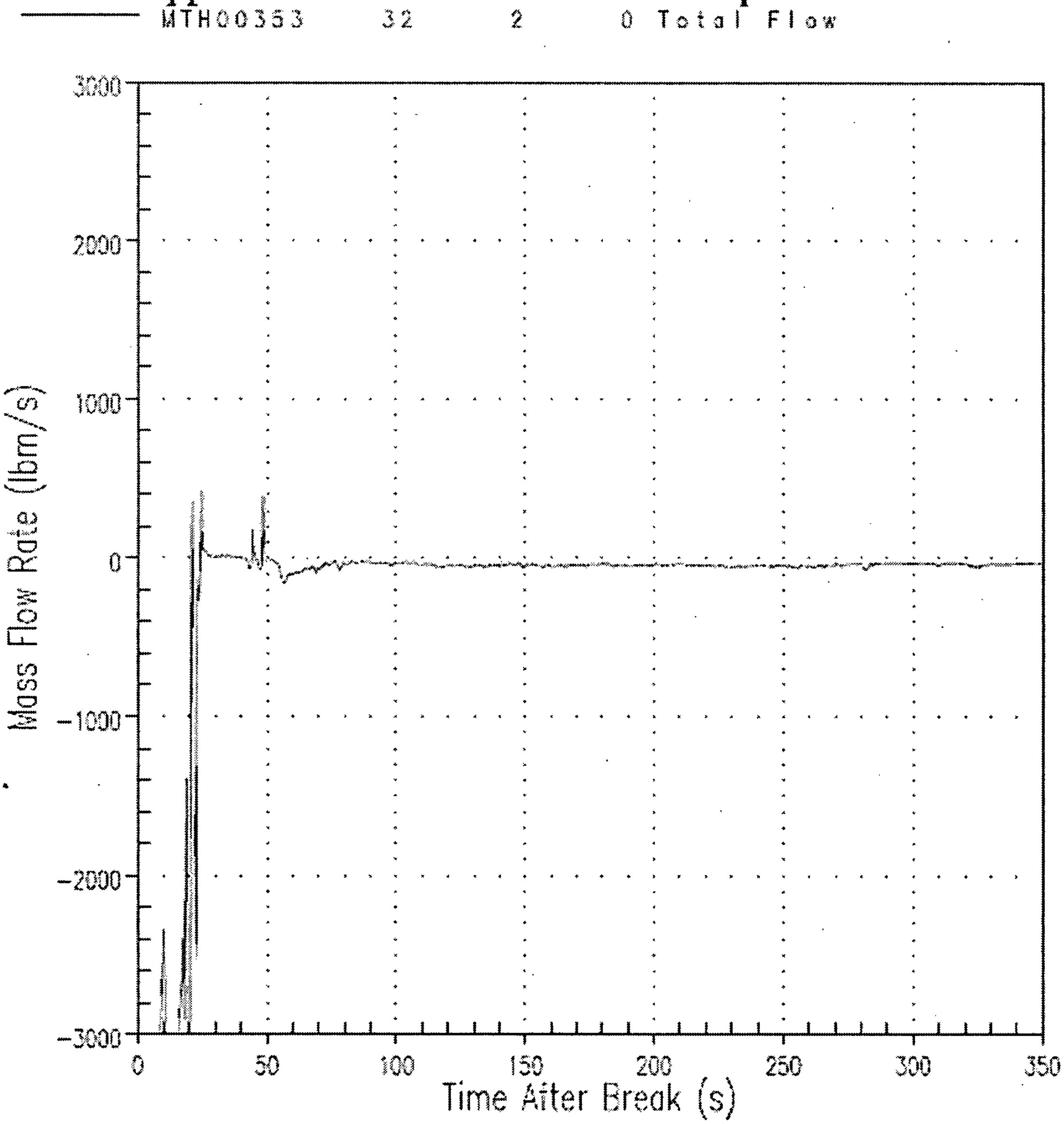
PTN Limiting PCT Case

Figure c69: Total Flow from Intact Loop Downcomer Channel 78 Upper Plenum - Section 6 - at Top of Section



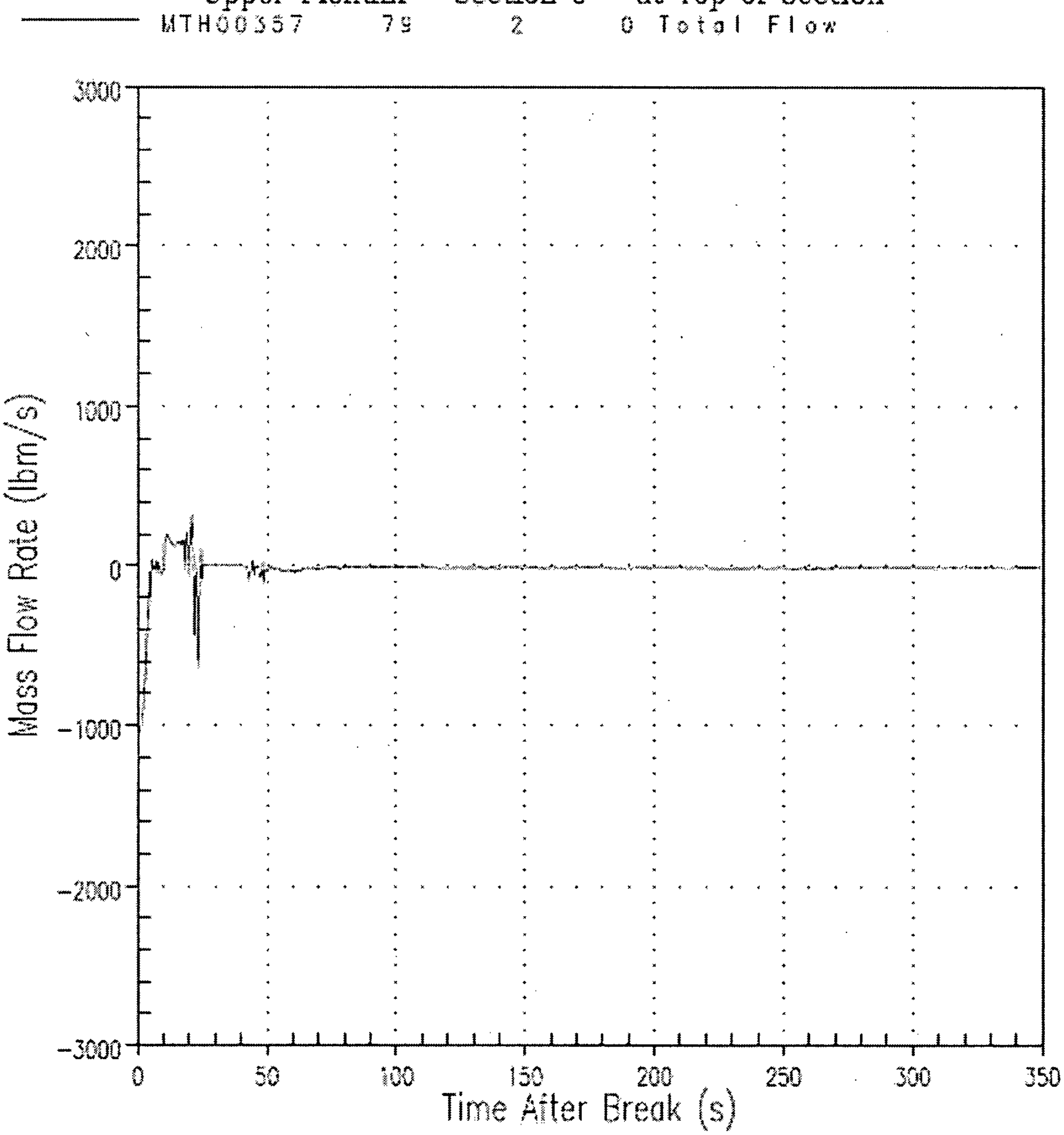
PTN Limiting PCT Case

Figure c70: Total Flow from Broken Loop Downcomer Channel 32
Upper Plenum – Section 6 – at Top of Section



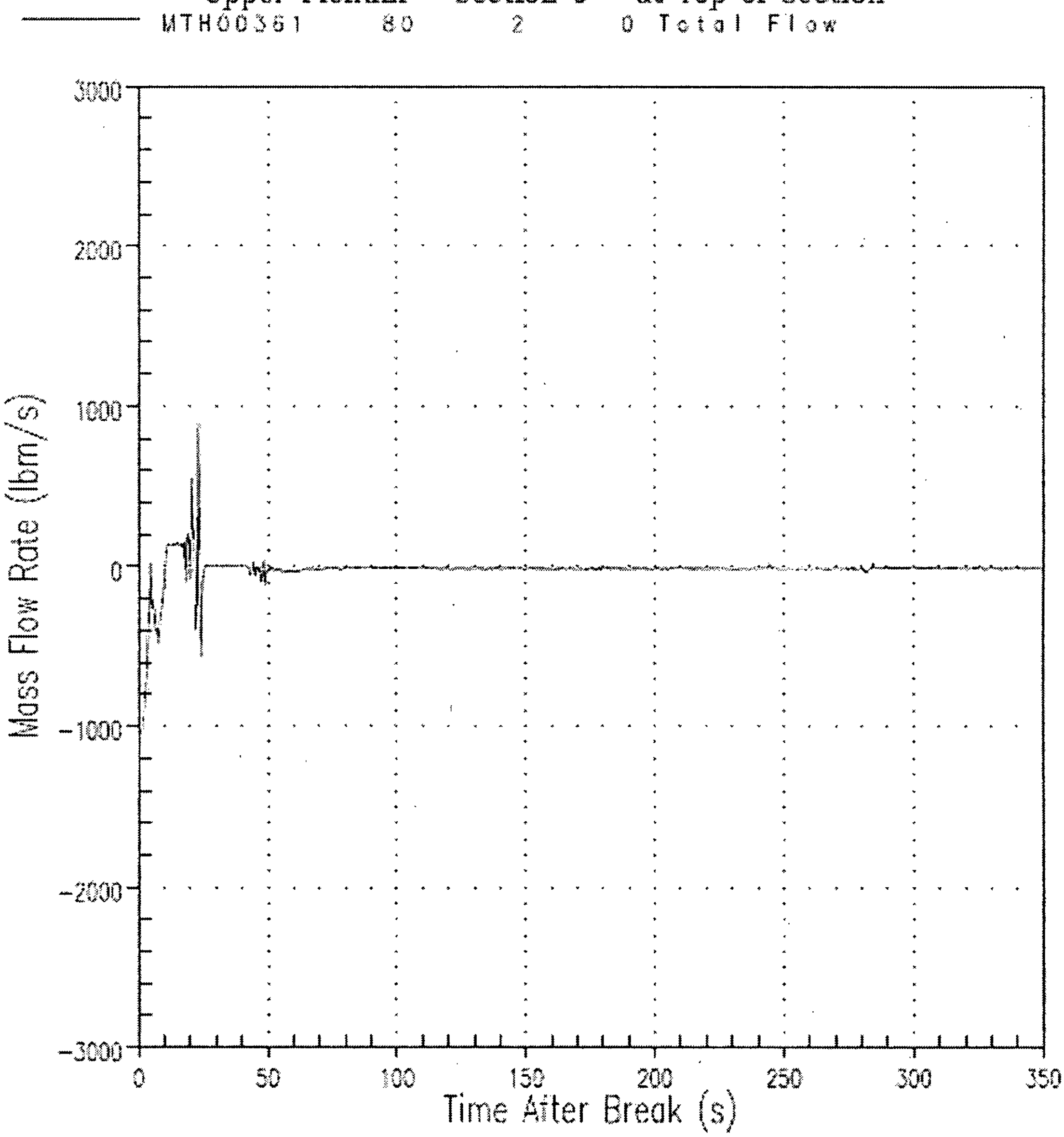
PTN Limiting PCT Case

Figure c71: Total Flow from Adjacen to Broken Loop Downcomer Channel 79
Upper Plenum – Section 6 – at Top of Section



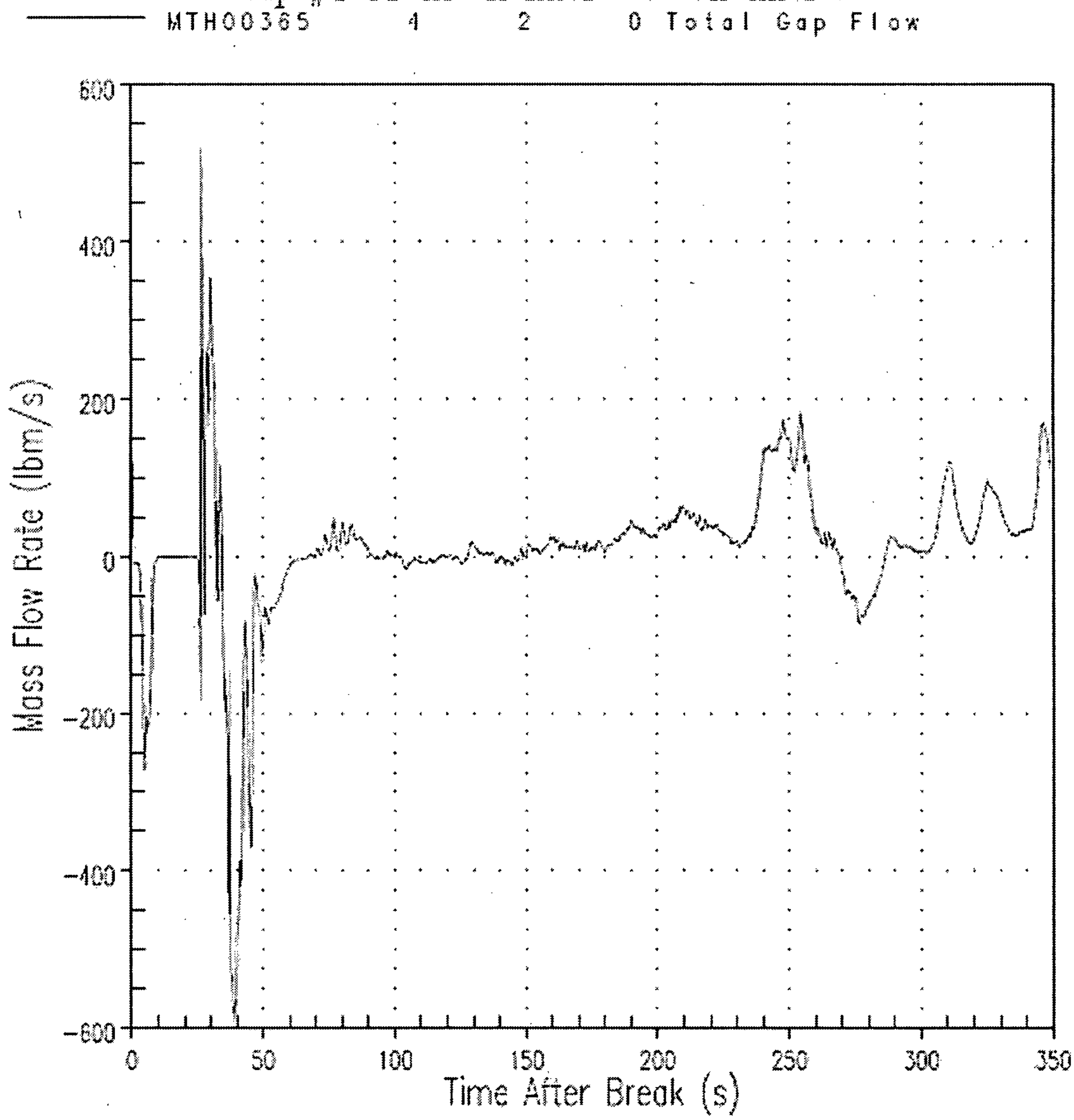
PTN Limiting PCT Case

Figure c72: Total Flow from Adjacen to Broken Loop Downcomer Channel 80
Upper Plenum - Section 6 - at Top of Section



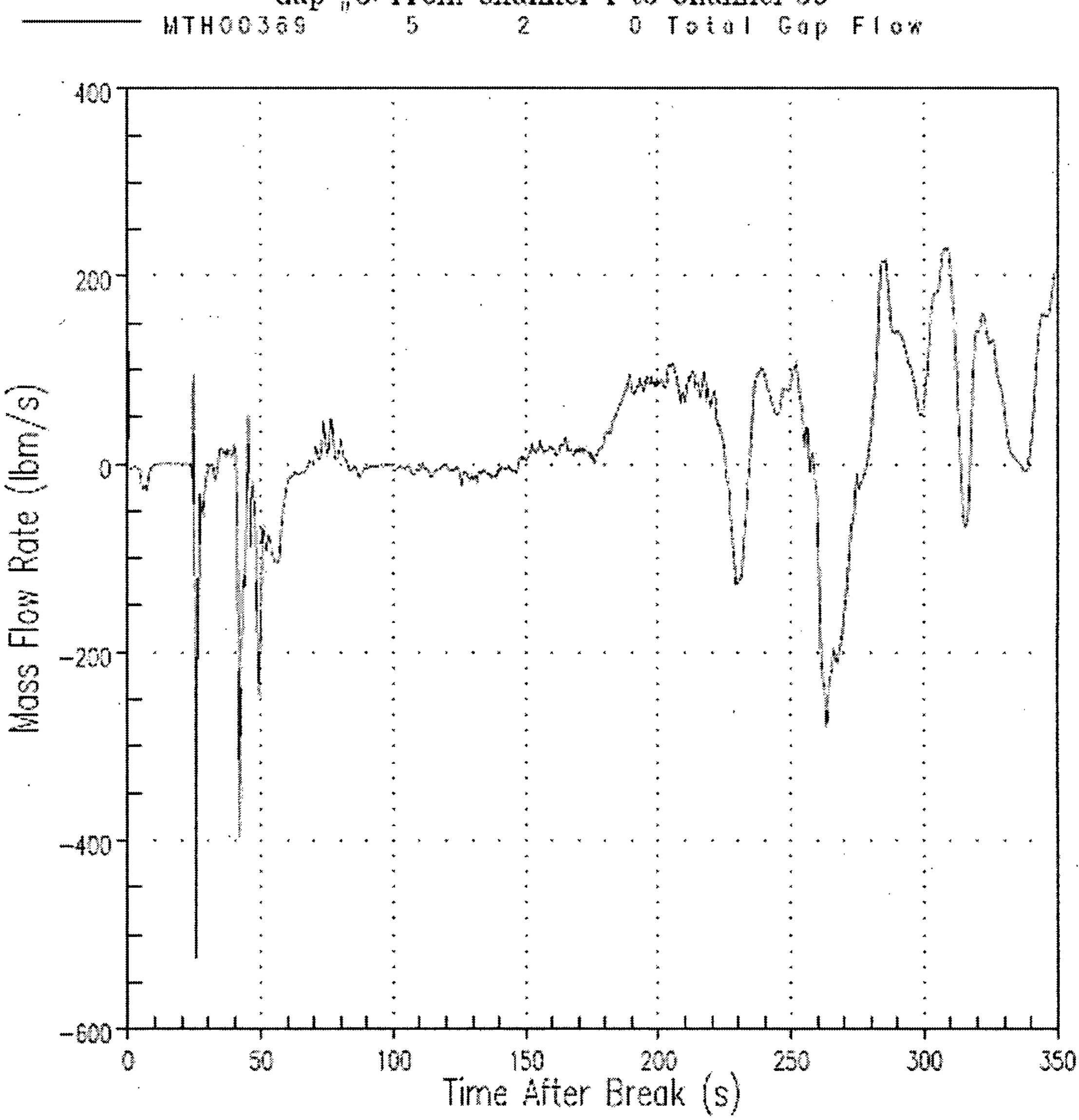
PTN Limiting PCT Case

Figure d1: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum – Section 2 – at Level 2
Gap #4: From Channel 2 to Channel 52



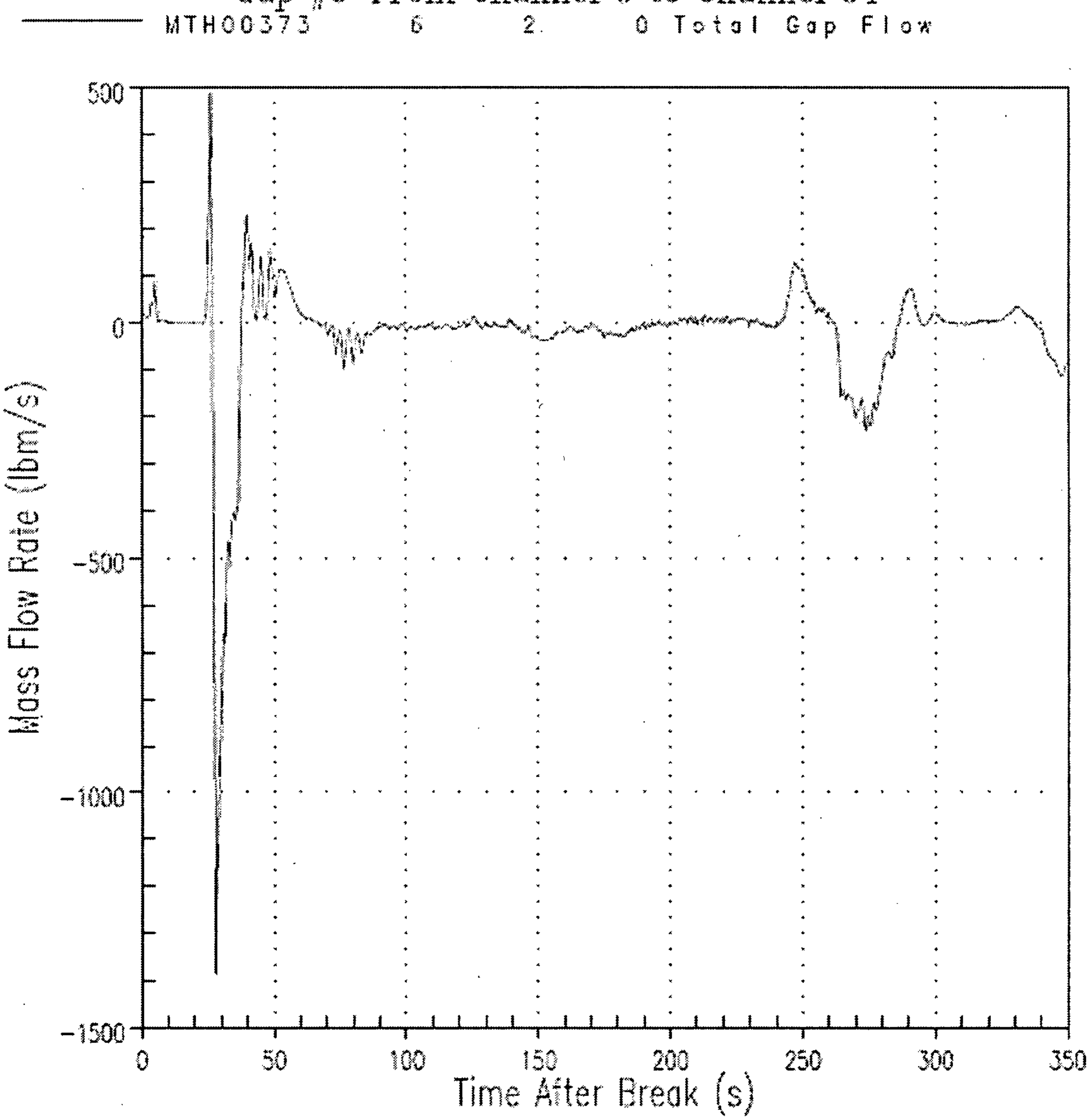
PTN Limiting PCT Case

Figure d2: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Lower Plenum – Section 2 – at Level 2
Gap #5: From Channel 4 to Channel 56



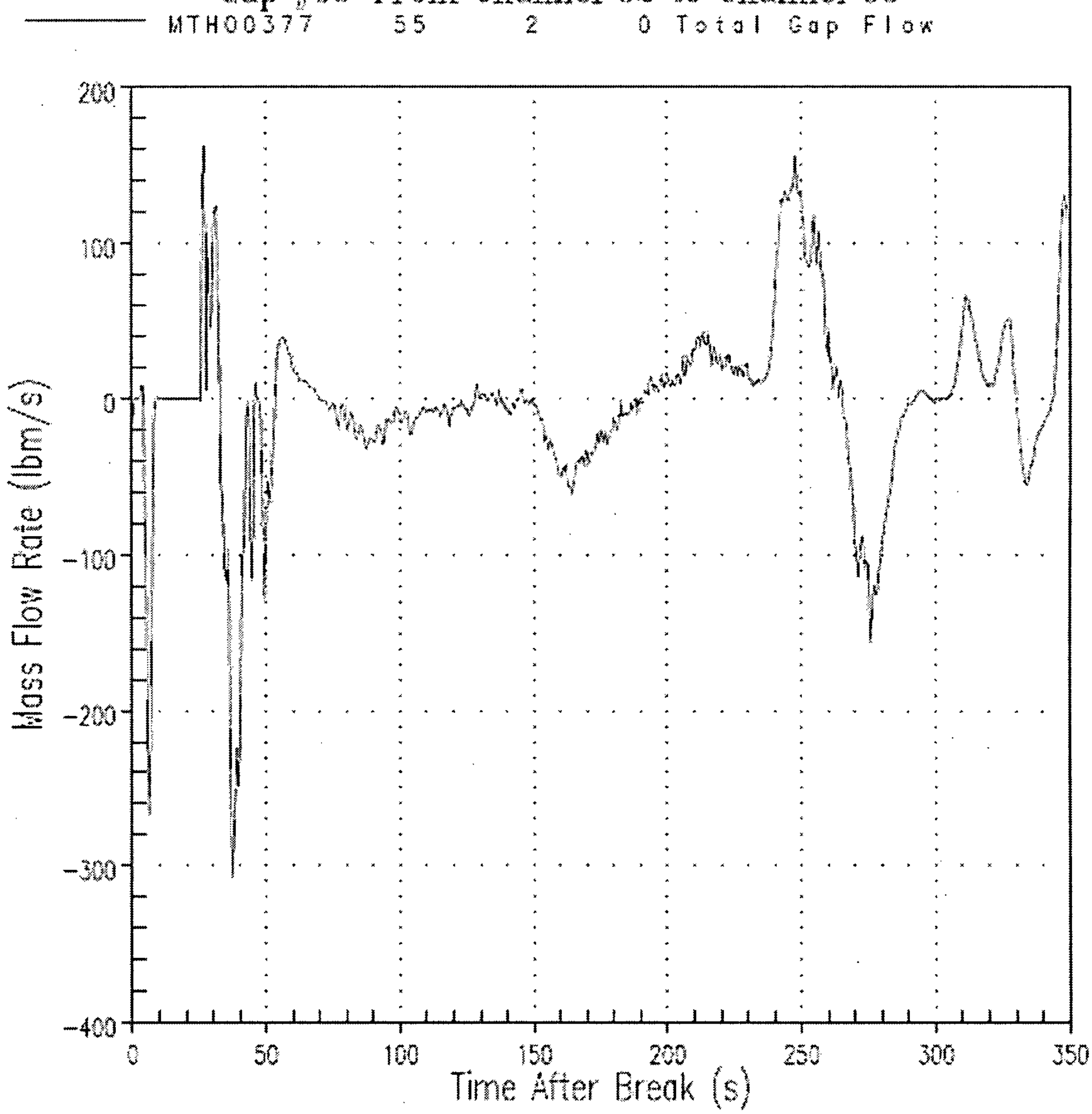
PTN Limiting PCT Case

Figure d3: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum – Section 2 – at Level 2
Gap #6: From Channel 3 to Channel 54



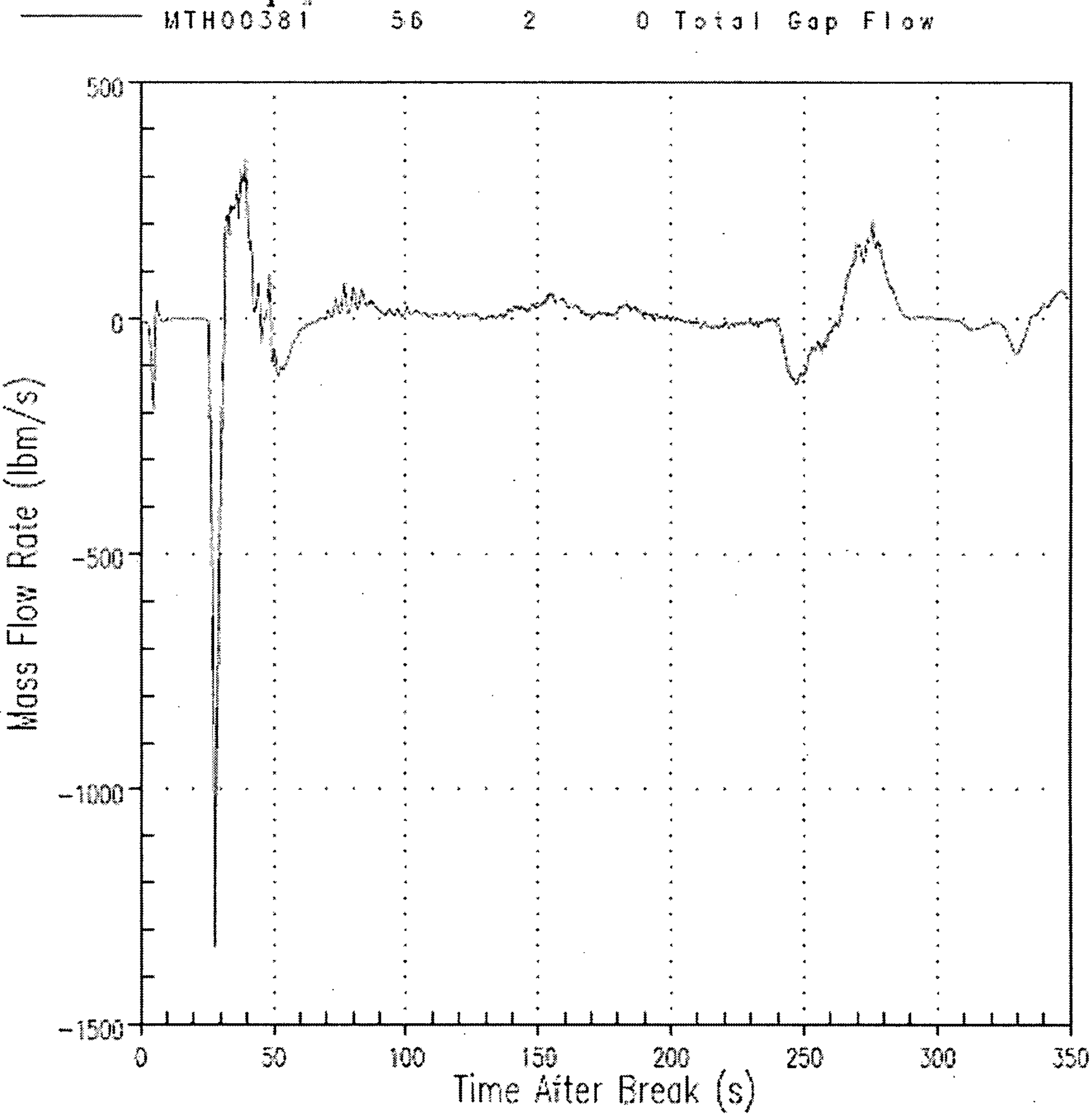
PTN Limiting PCT Case

**Figure d4: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum – Section 2 – at Level 2
Gap #55: From Channel 52 to Channel 53**



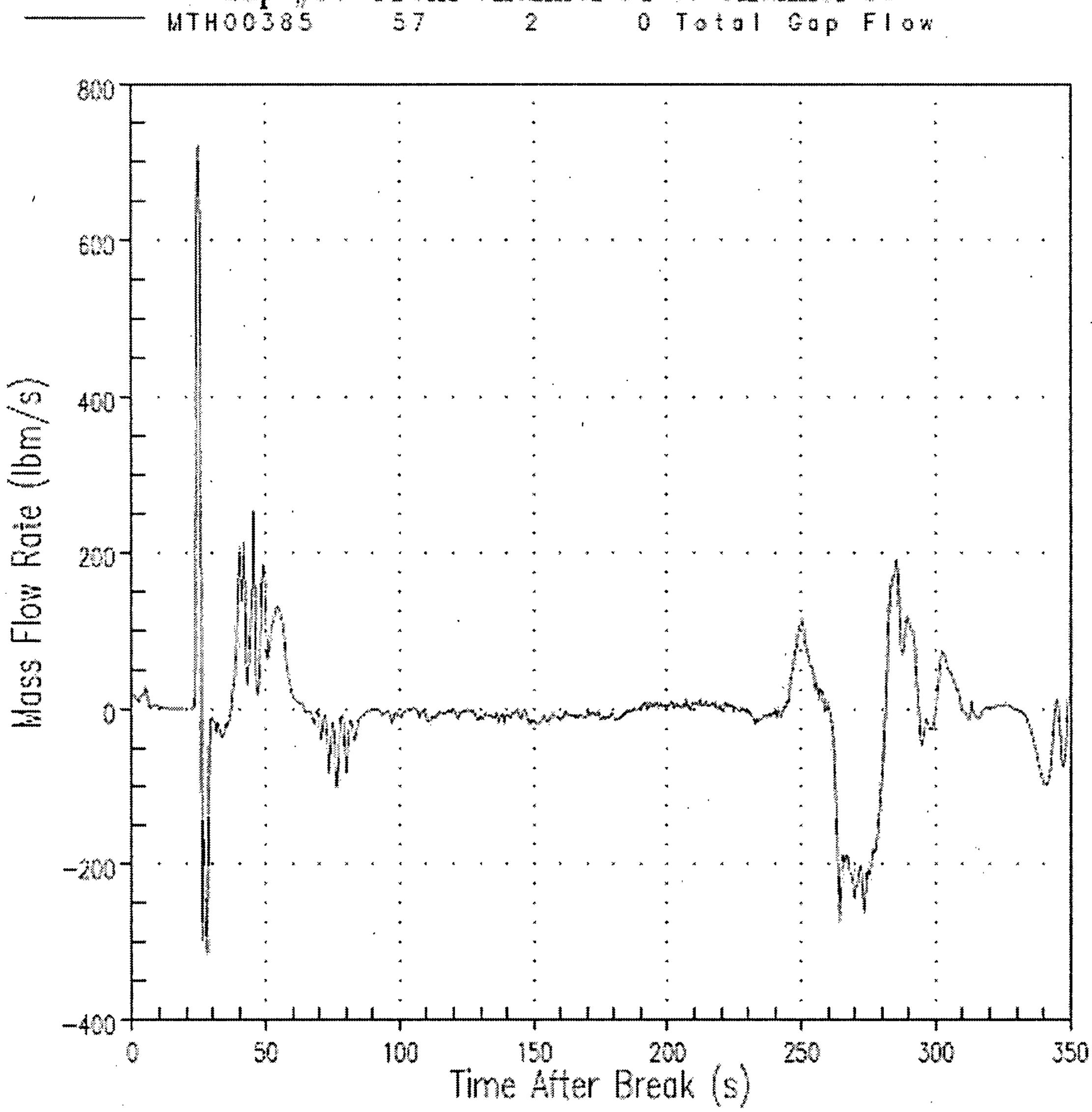
PTN Limiting PCT Case

Figure d5: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum - Section 2 - at Level 2
Gap #56: From Channel 3 to Channel 53



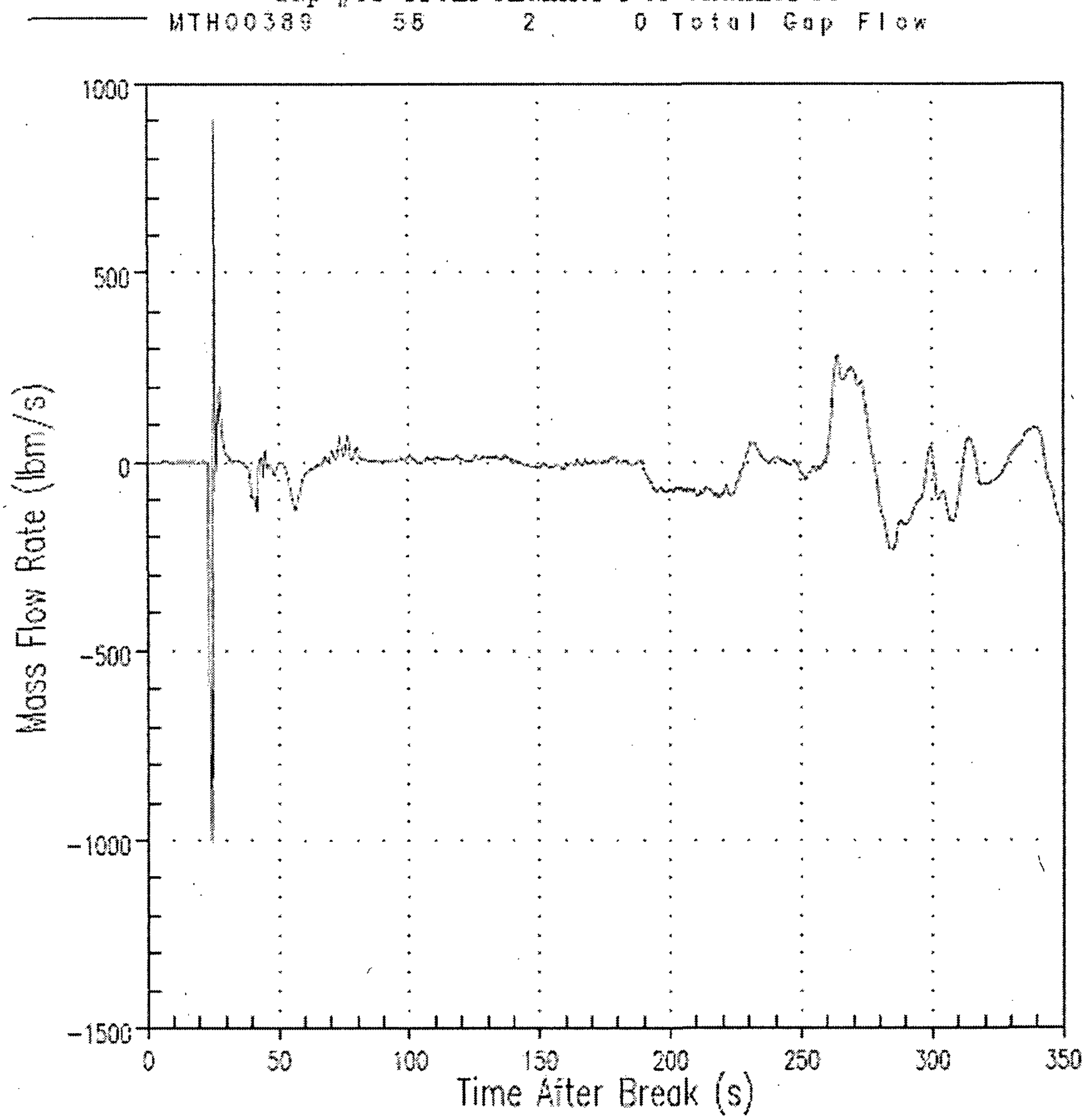
PTN Limiting PCT Case

Figure d6: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum - Section 2 - at Level 2
Gap #57: From Channel 54 to Channel 55



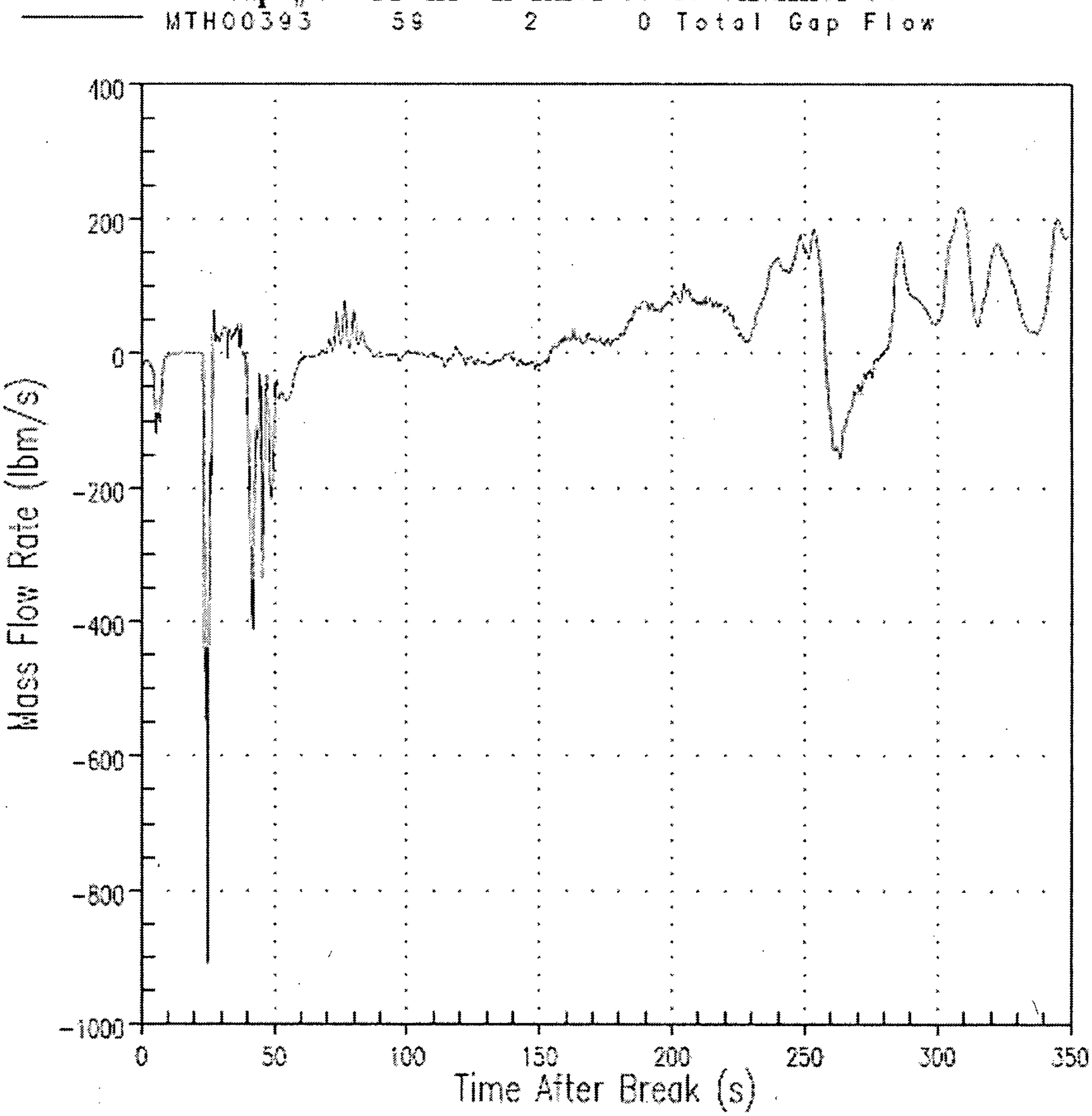
PTN Limiting PCT Case

Figure d7: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Lower Plenum – Section 2 – at Level 2
Gap #58: From Channel 4 to Channel 55



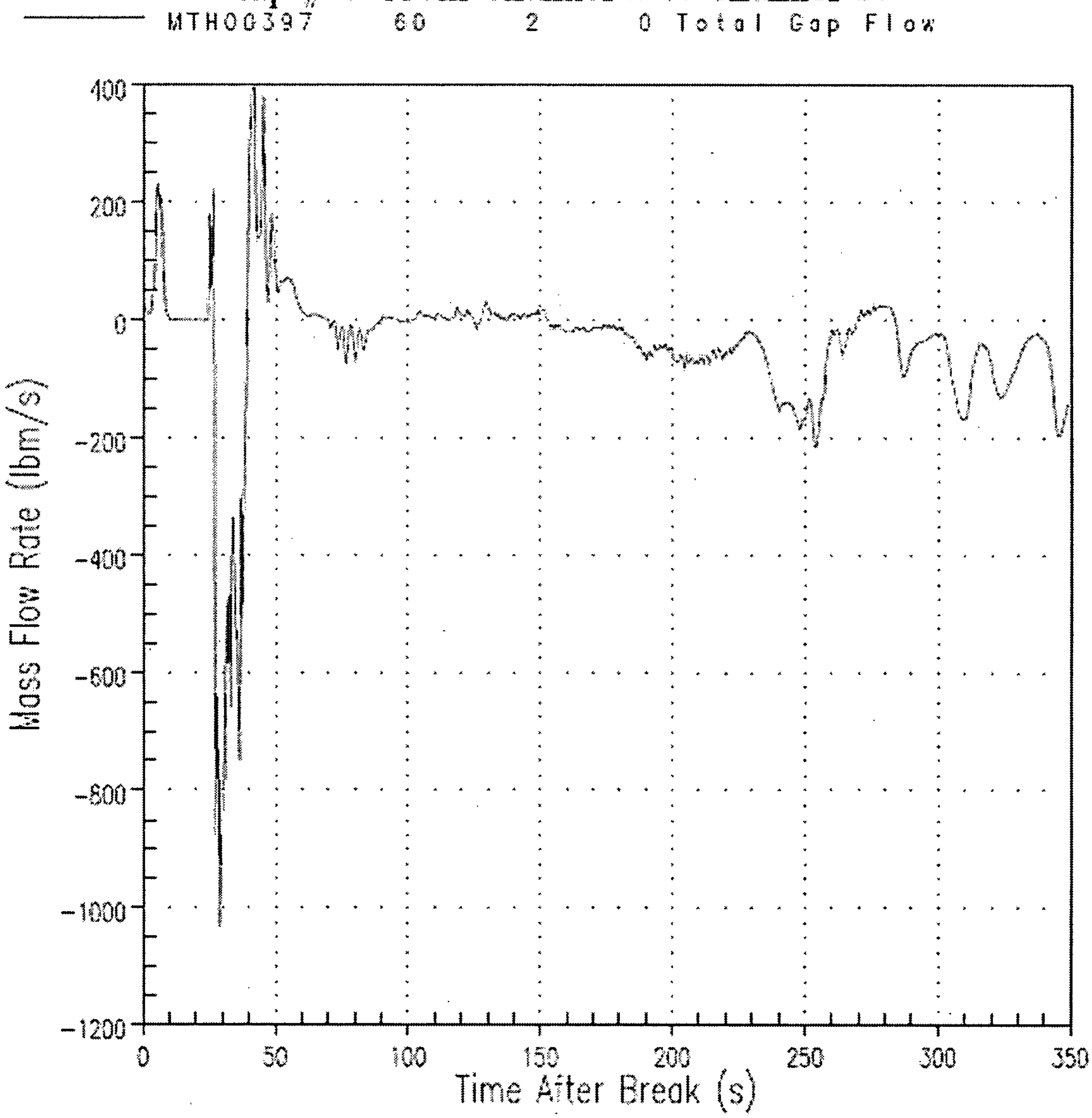
PTN Limiting PCT Case

Figure d8: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum - Section 2 - at Level 2
Gap #59: From Channel 56 to Channel 57



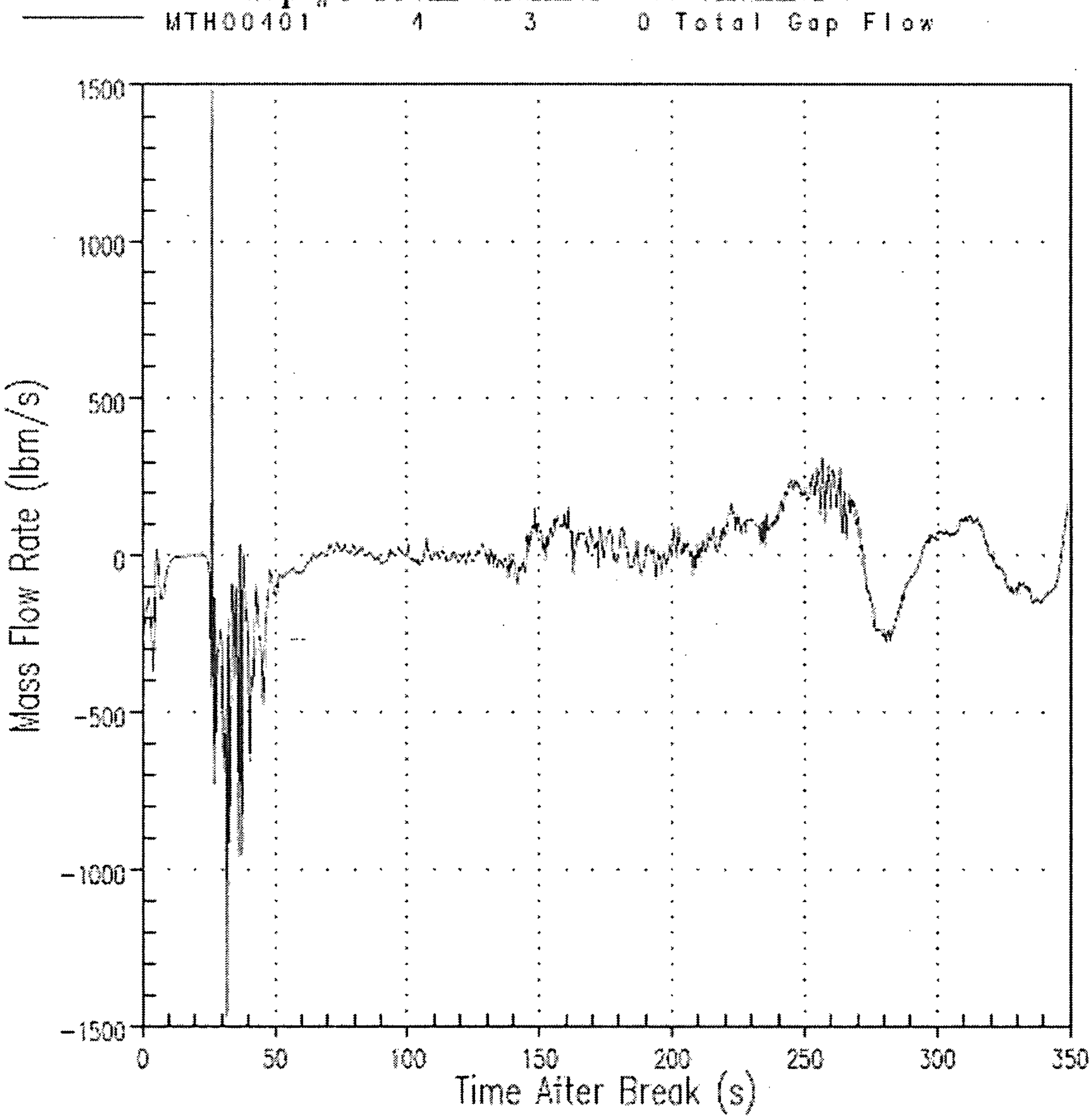
PTN Limiting PCT Case

Figure d9: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum – Section 2 – at Level 2
Gap #60: From Channel 2 to Channel 57



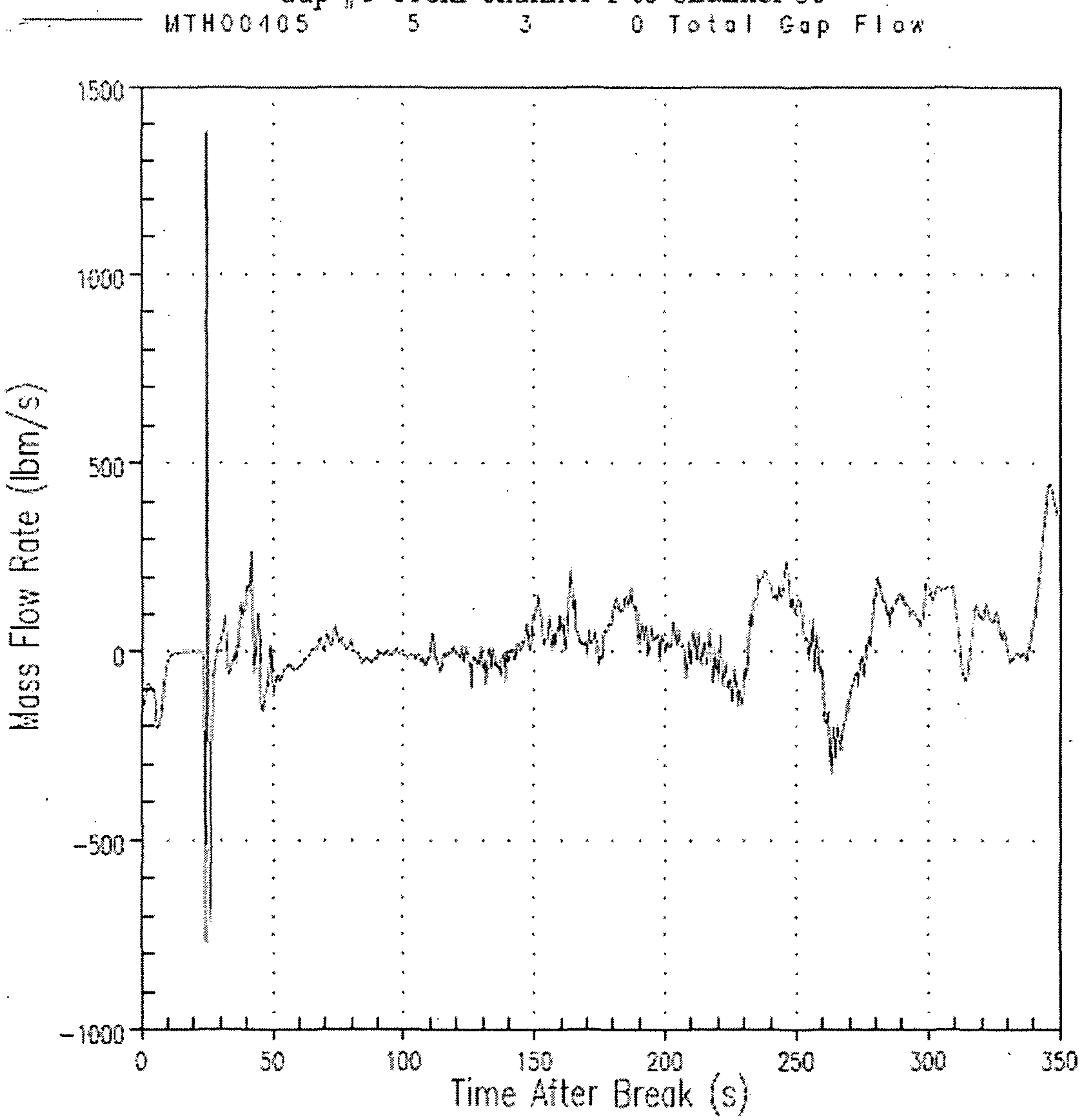
PTN Limiting PCT Case

Figure d10: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum – Section 2 – at Level 3
Gap #4: From Channel 2 to Channel 52



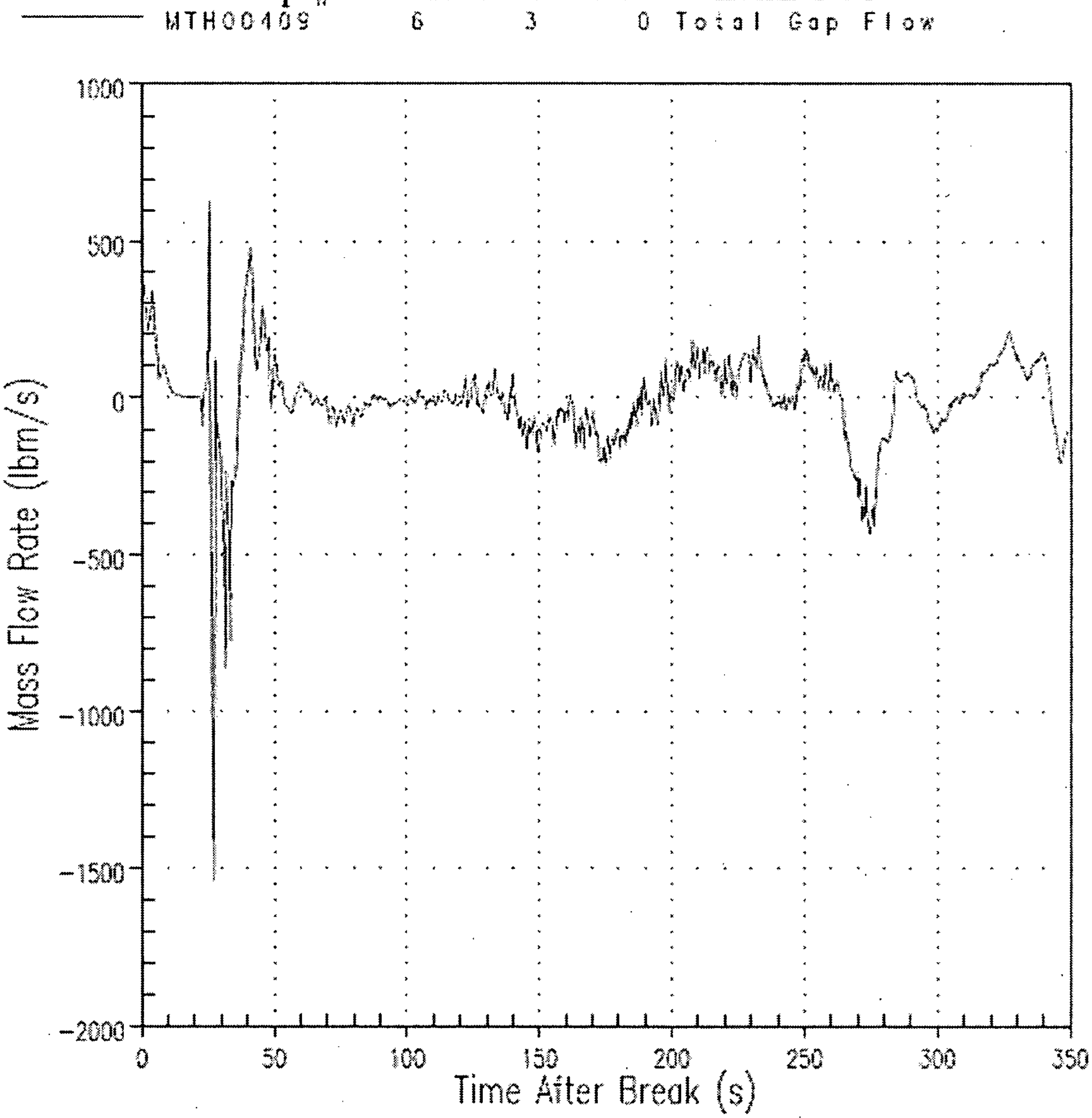
PTN Limiting PCT Case

Figure d11: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Lower Plenum – Section 2 – at Level 3
Gap #5: From Channel 4 to Channel 56



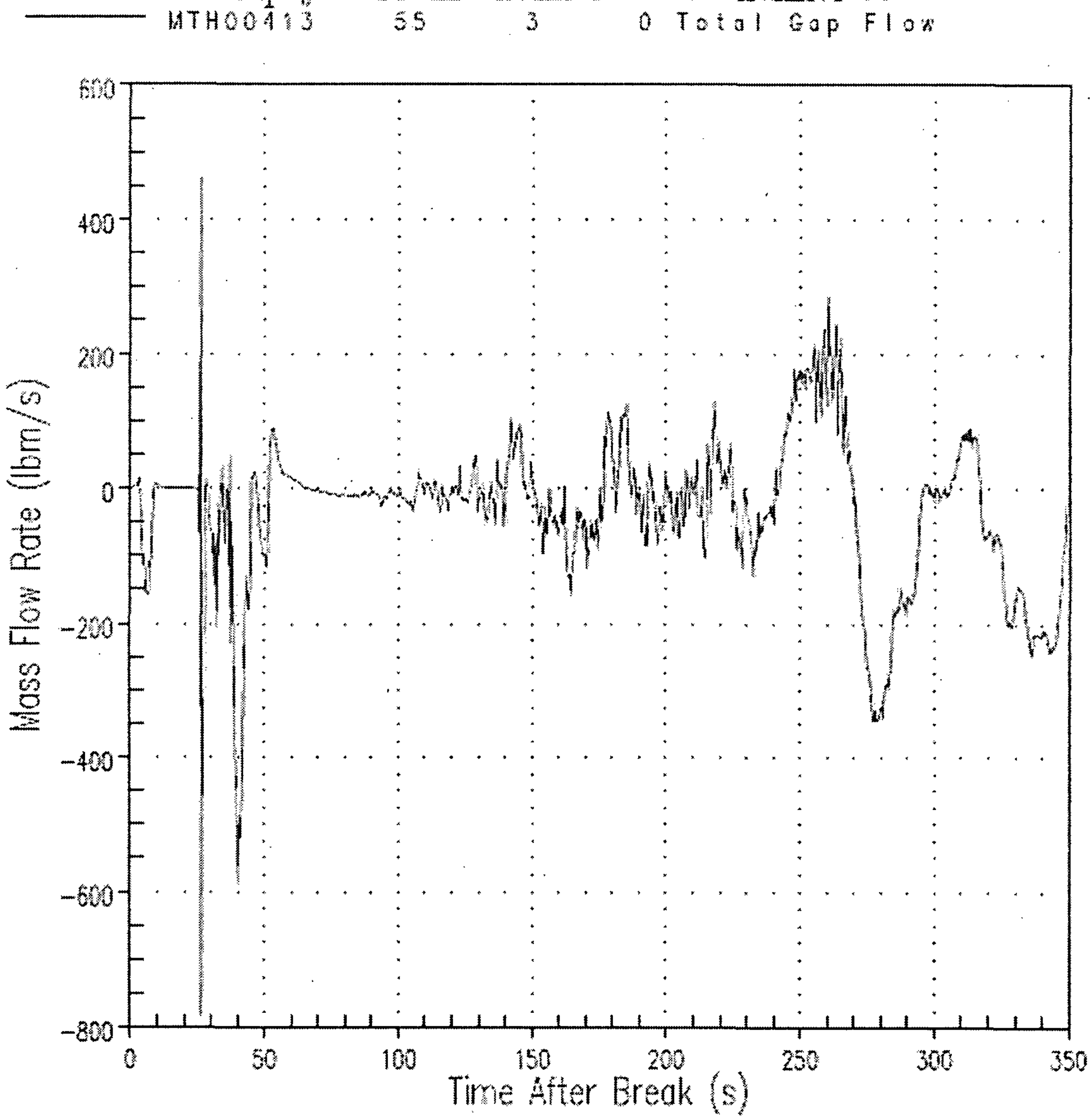
PTN Limiting PCT Case

Figure d12: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum - Section 2 - at Level 3
Gap #6: From Channel 3 to Channel 54



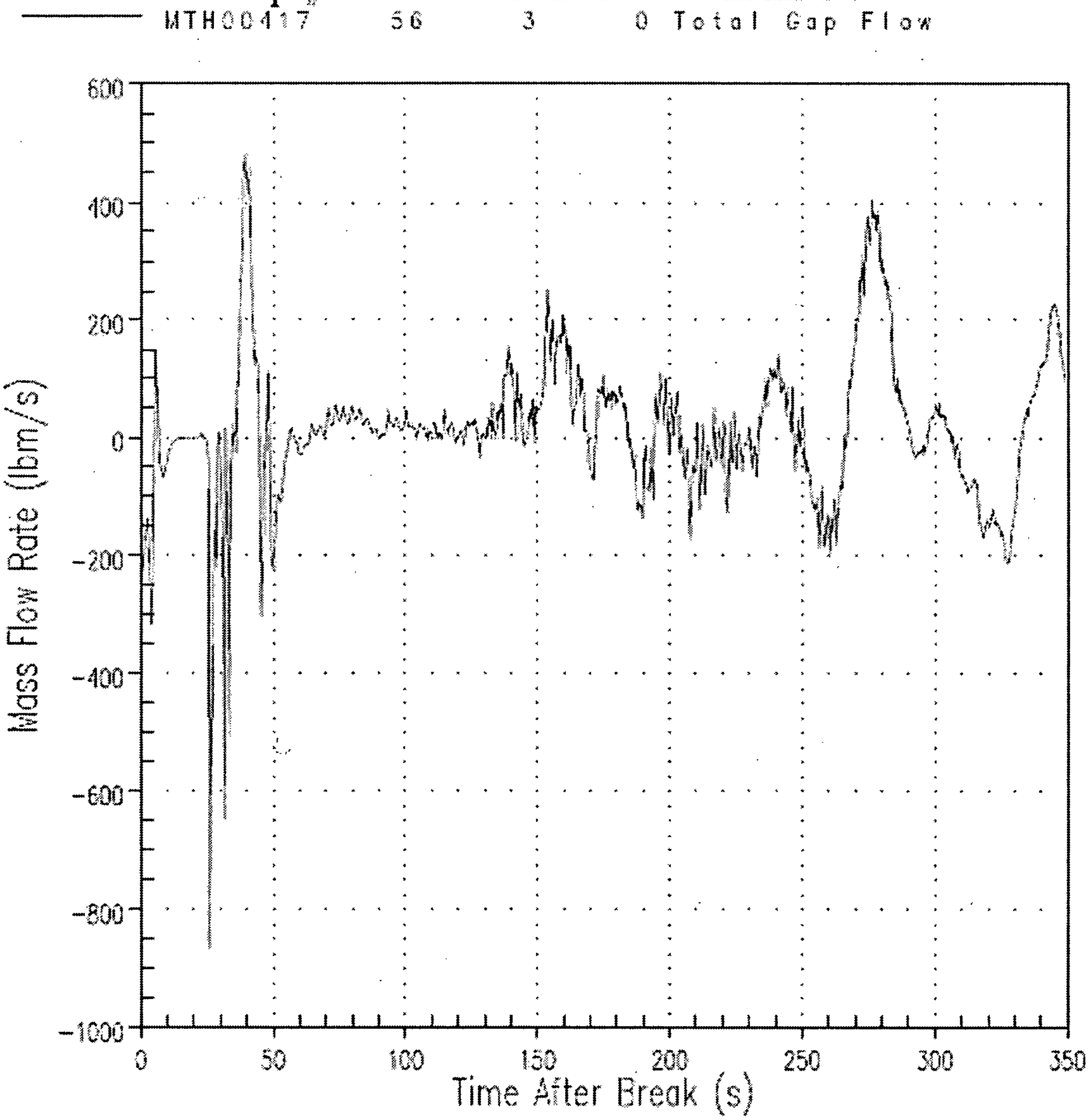
PTN Limiting PCT Case

Figure d13: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum - Section 2 - at Level 3
Gap #55: From Channel 52 to Channel 53



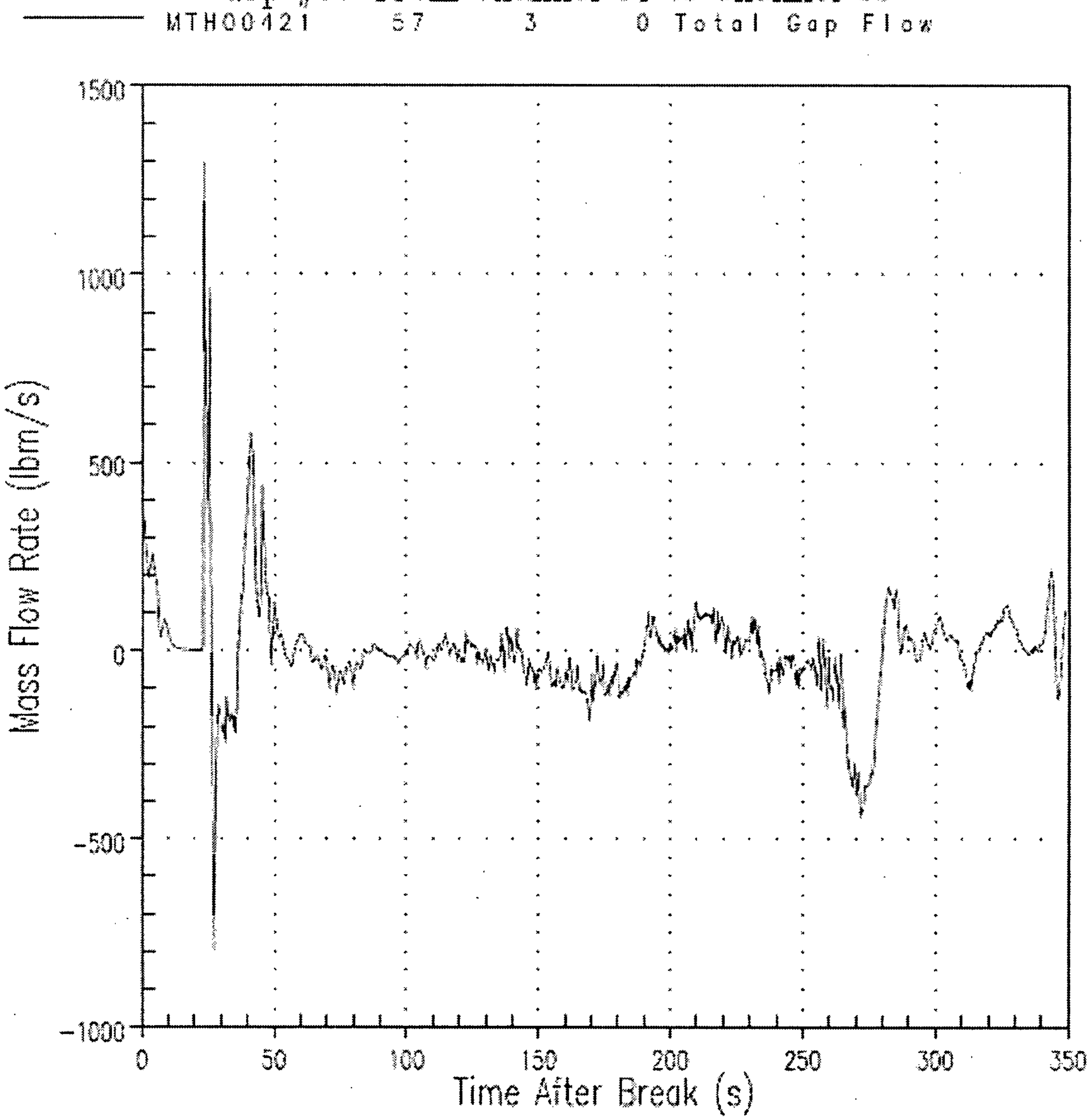
PTN Limiting PCT Case

Figure d14: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum – Section 2 – at Level 3
Gap #56: From Channel 3 to Channel 53



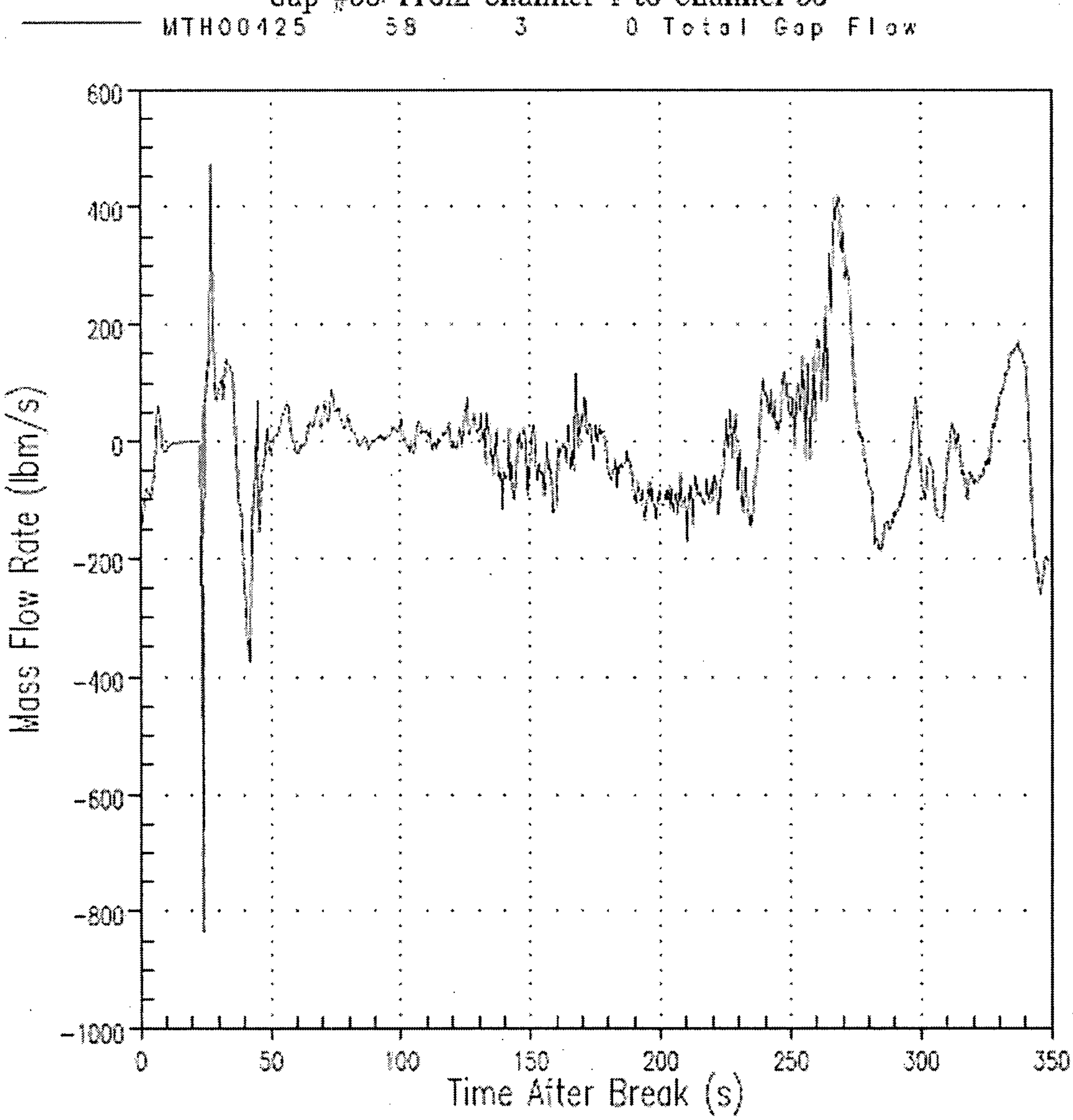
PTN Limiting PCT Case

Figure d15: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum - Section 2 - at Level 3
Gap #57: From Channel 54 to Channel 55



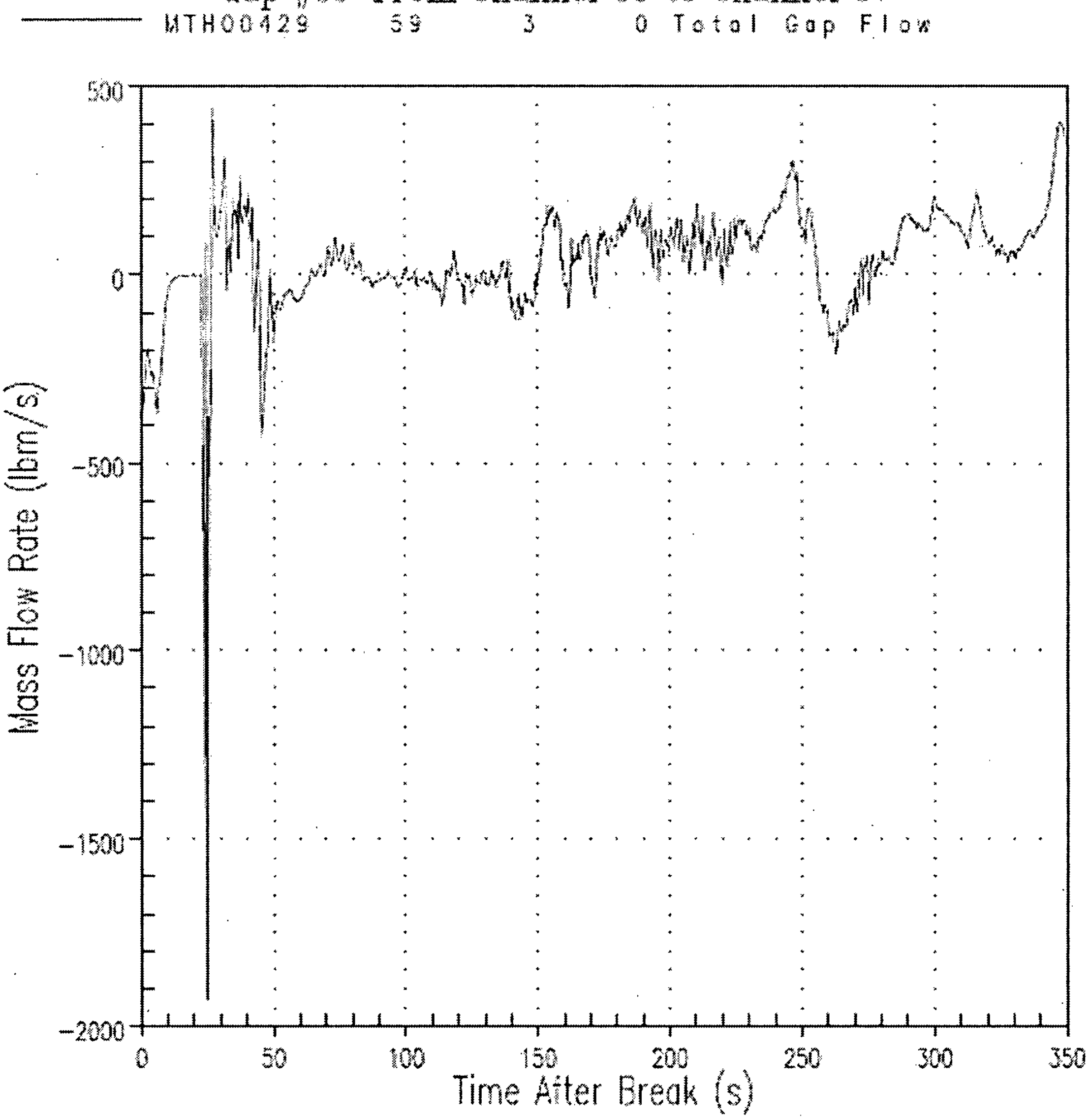
PTN Limiting PCT Case

Figure d16: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Lower Plenum - Section 2 - at Level 3
Gap #58: From Channel 4 to Channel 55



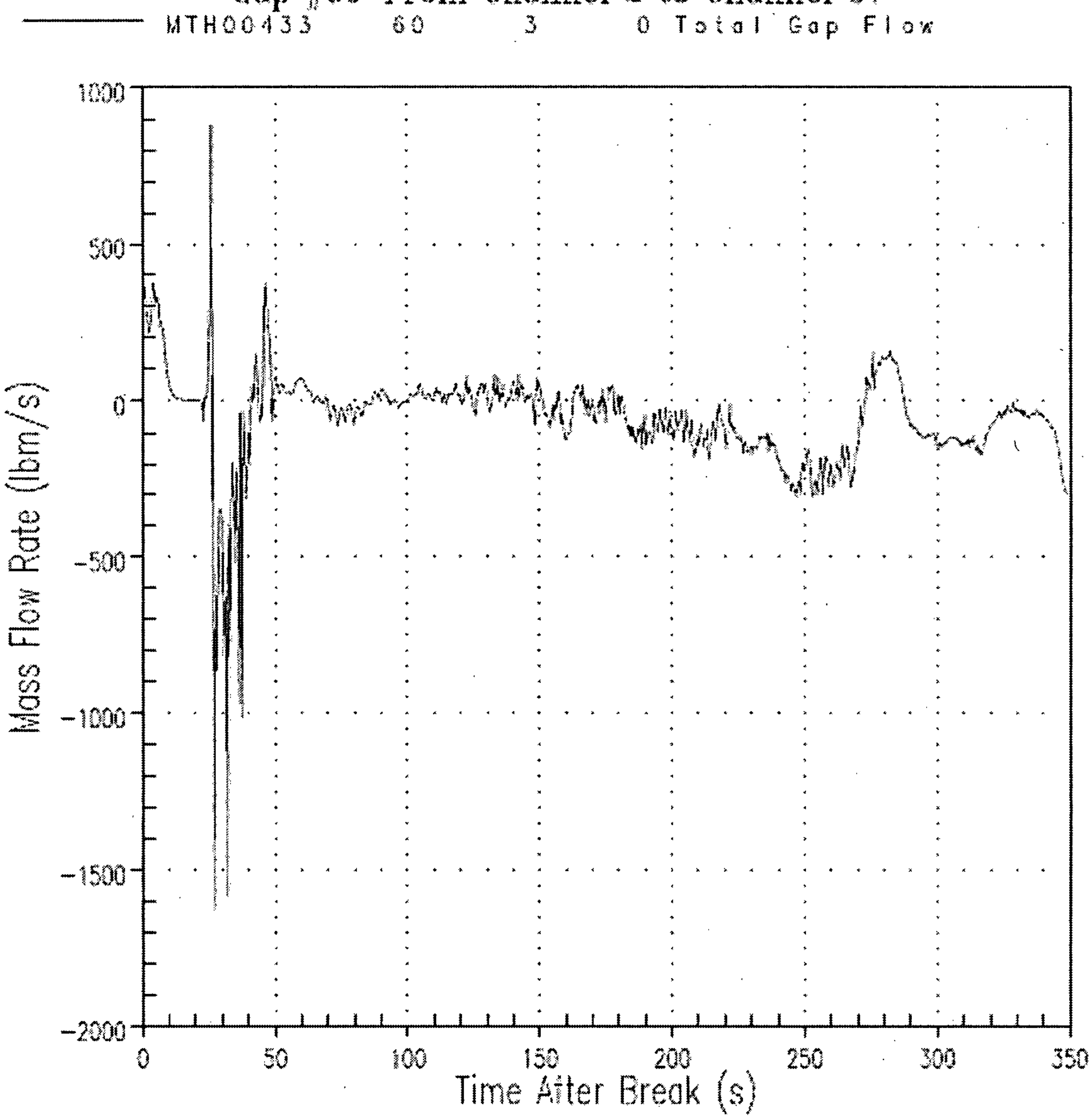
PTN Limiting PCT Case

Figure d17: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum - Section 2 - at Level 3
Gap #59: From Channel 56 to Channel 57



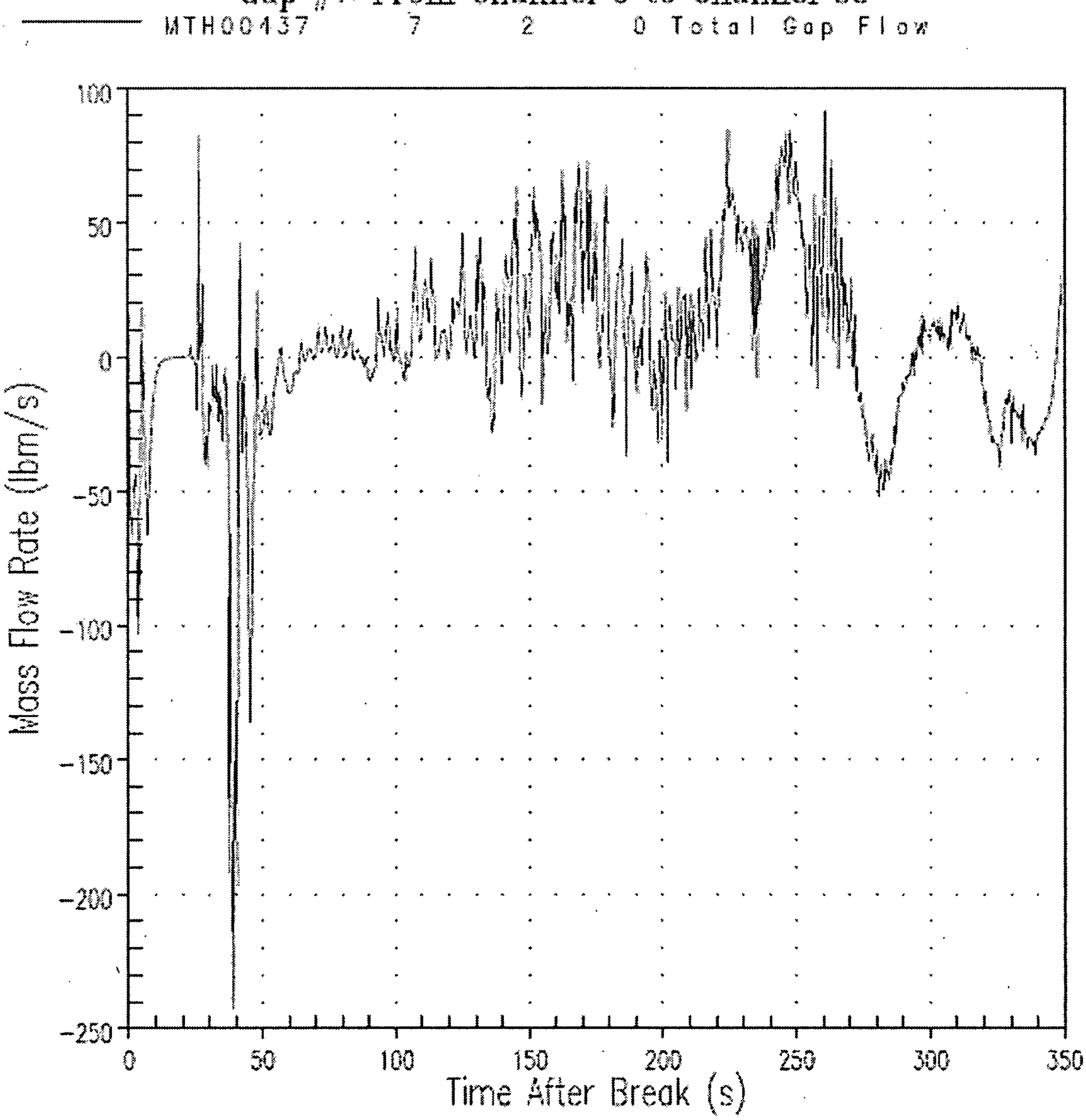
PTN Limiting PCT Case

Figure d18: Total Gap Flow between Intact Loop Downcomer Channels
Lower Plenum - Section 2 - at Level 3
Gap #60: From Channel 2 to Channel 57



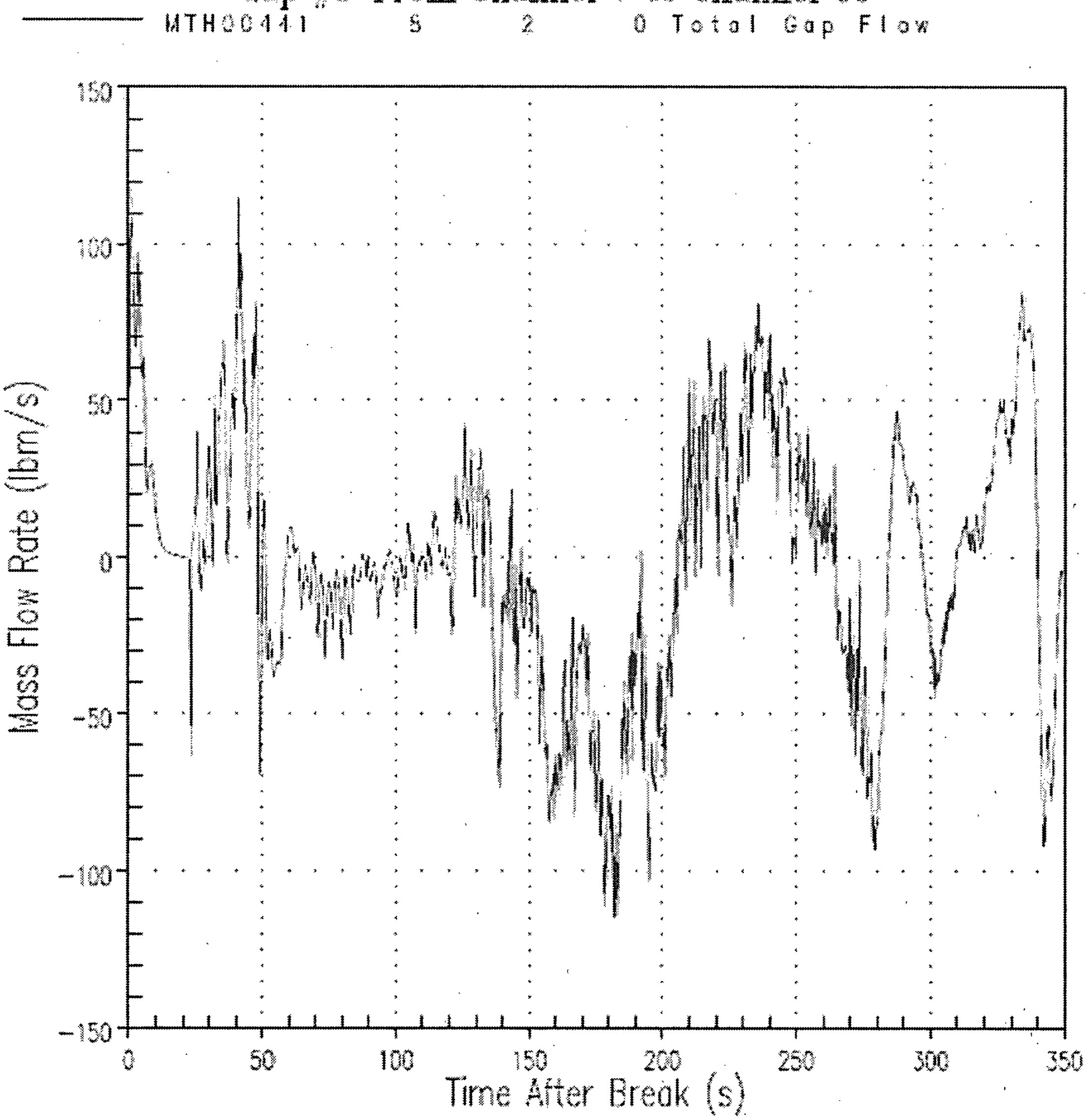
PTN Limiting PCT Case

Figure d19: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Bottom of Section
Gap #7: From Channel 6 to Channel 58



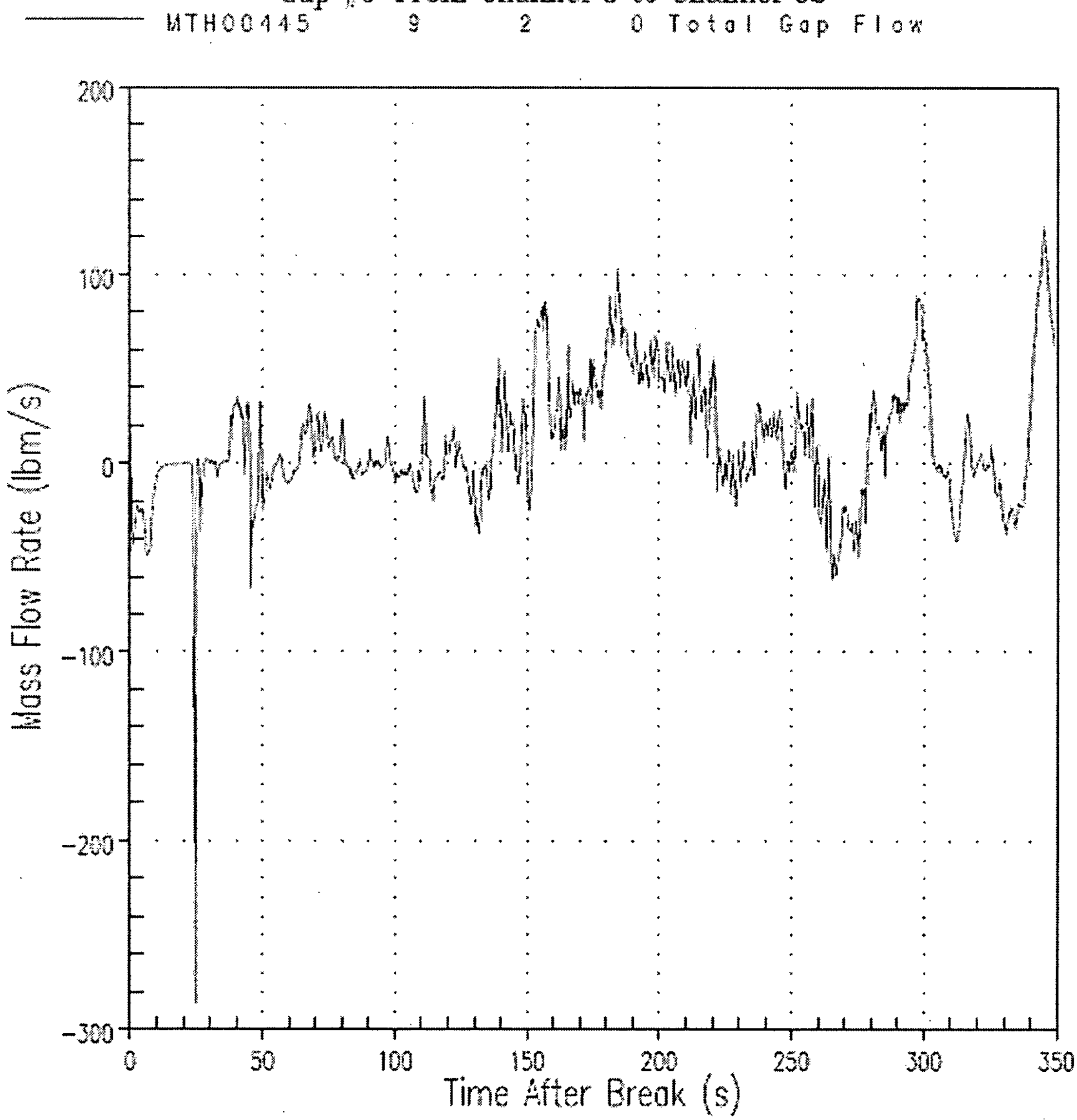
PTN Limiting PCT Case

Figure d20: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Bottom of Section
Gap #8: From Channel 7 to Channel 60



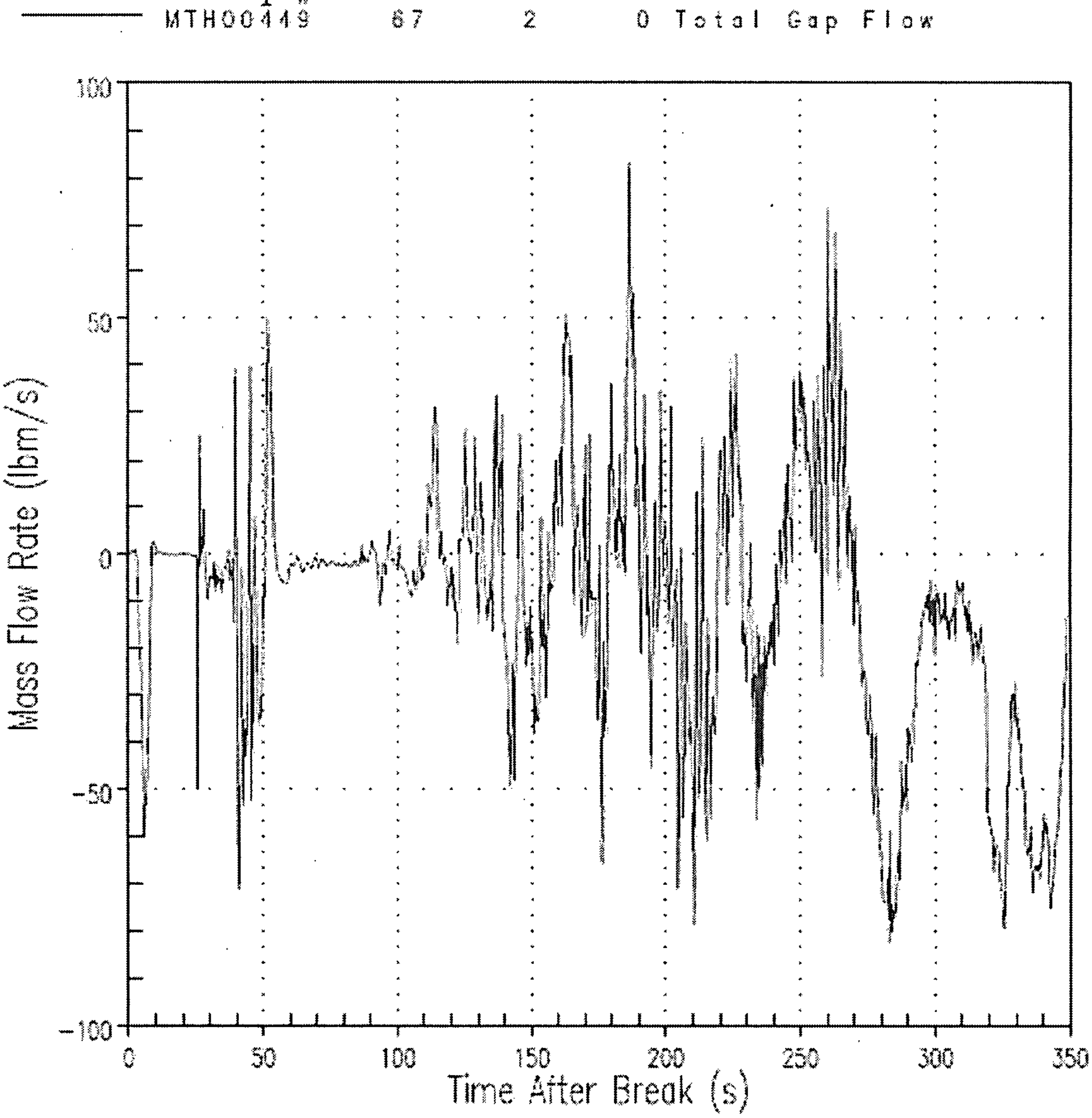
PTN Limiting PCT Case

Figure d21: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Core - Section 3 - at Bottom of Section
Gap #9: From Channel 8 to Channel 62



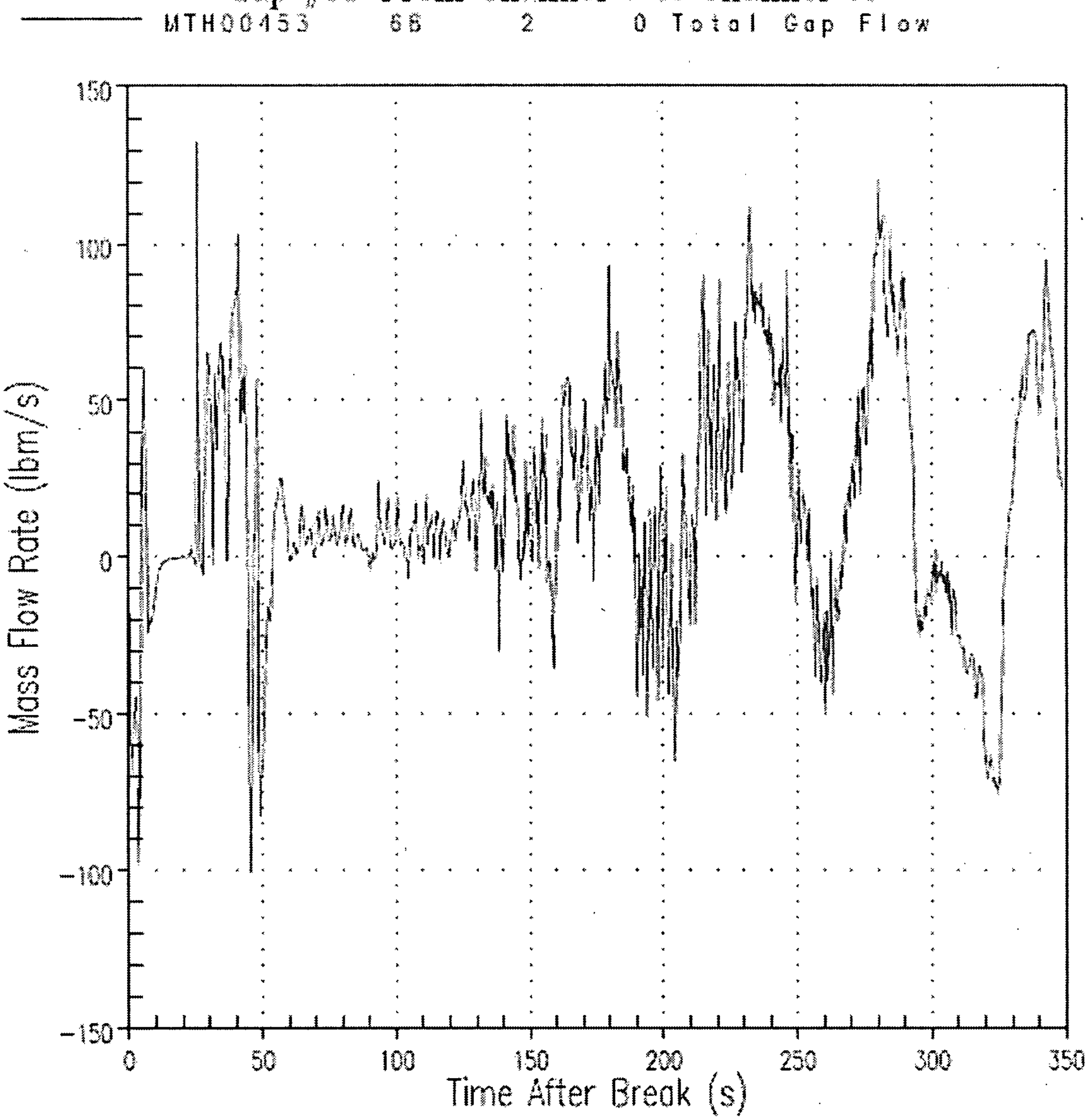
PTN Limiting PCT Case

Figure d22: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Bottom of Section
Gap #67: From Channel 58 to Channel 59



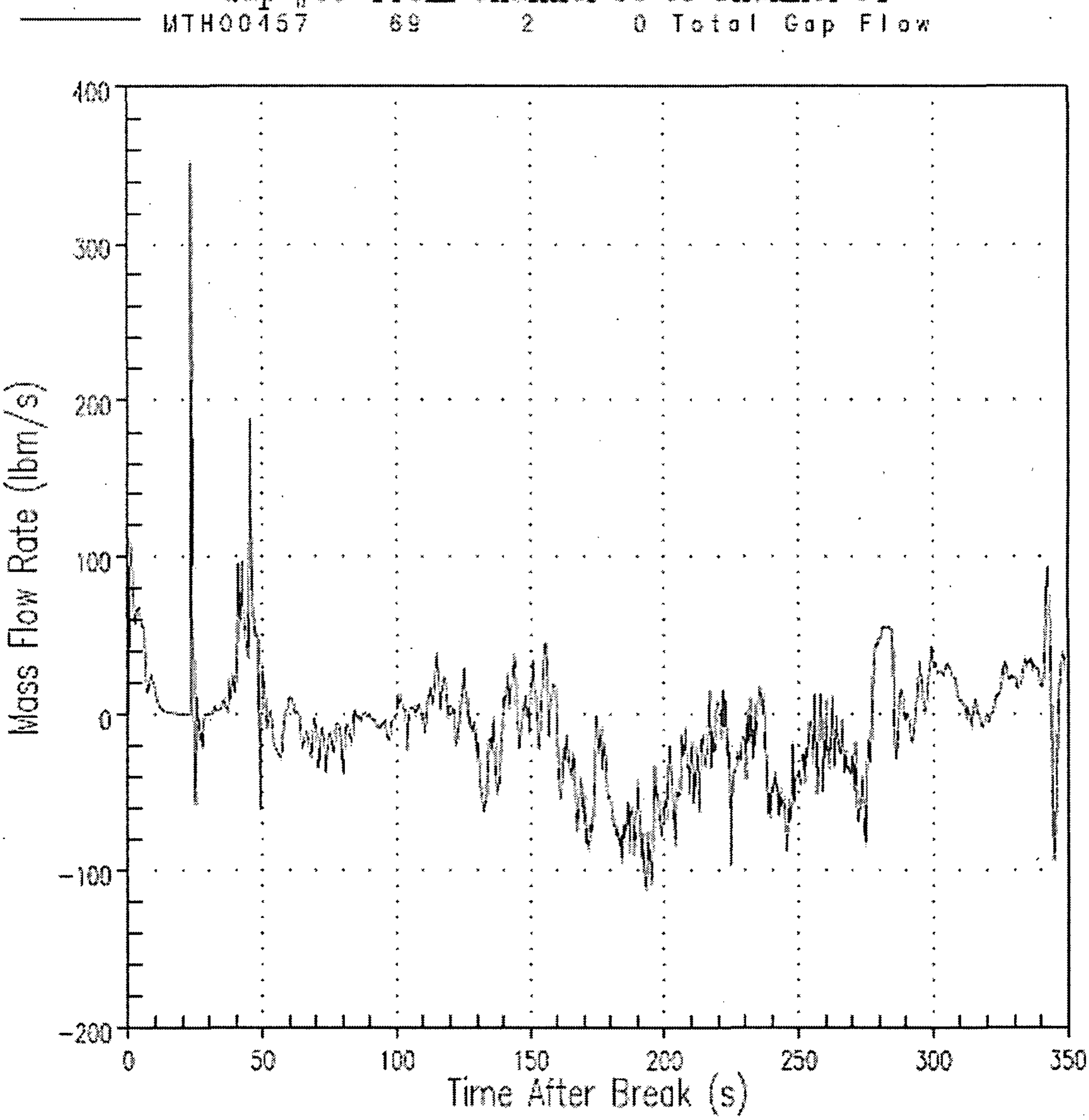
PTN Limiting PCT Case

**Figure d23: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Bottom of Section
Gap #68: From Channel 7 to Channel 59**



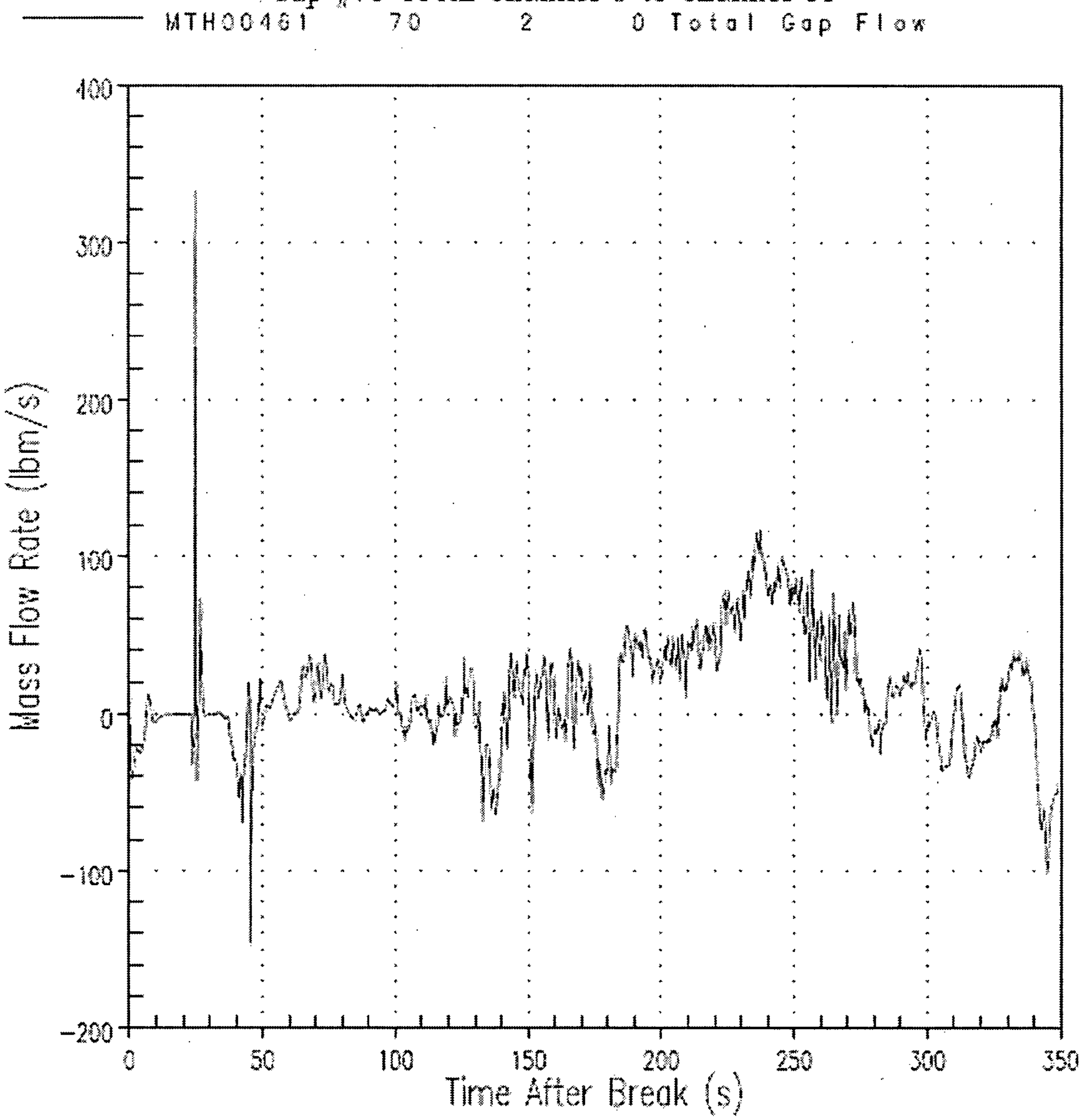
PTN Limiting PCT Case

Figure d24: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Bottom of Section
Gap #69: From Channel 60 to Channel 61



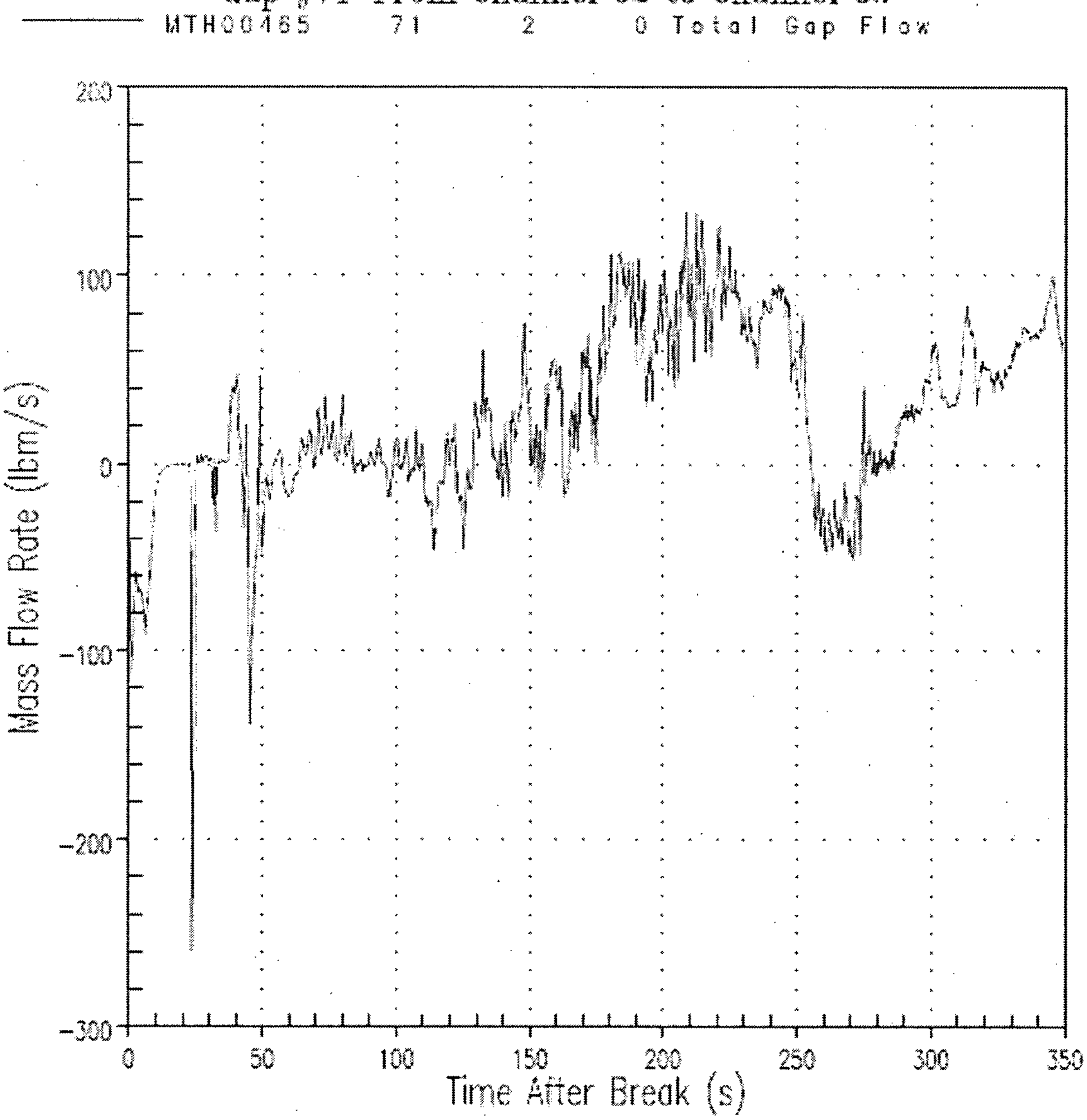
PTN Limiting PCT Case

Figure d25: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Core - Section 3 - at Bottom of Section
Gap #70: From Channel 8 to Channel 61



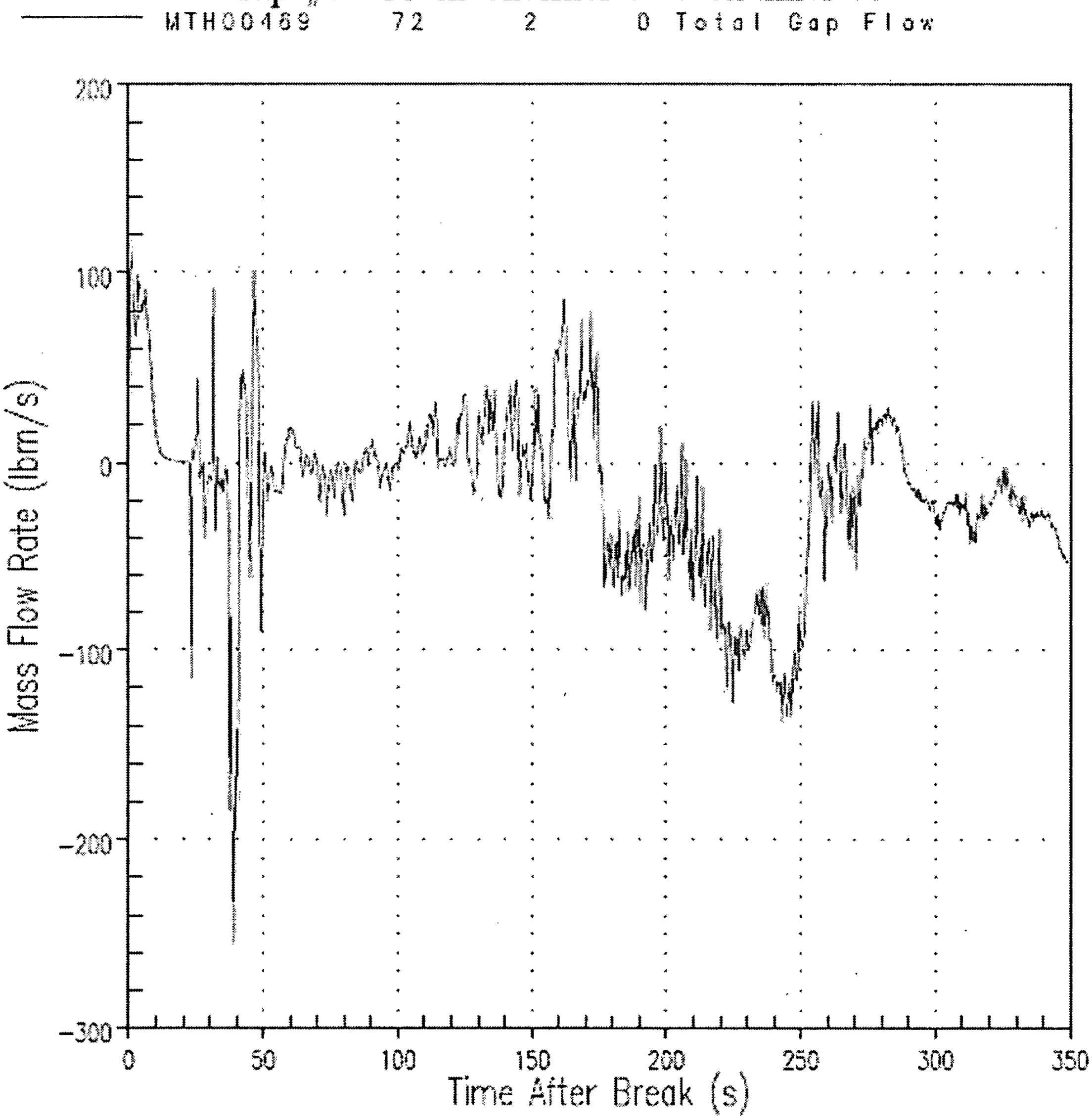
PTN Limiting PCT Case

Figure d26: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Bottom of Section
Gap #71: From Channel 62 to Channel 63



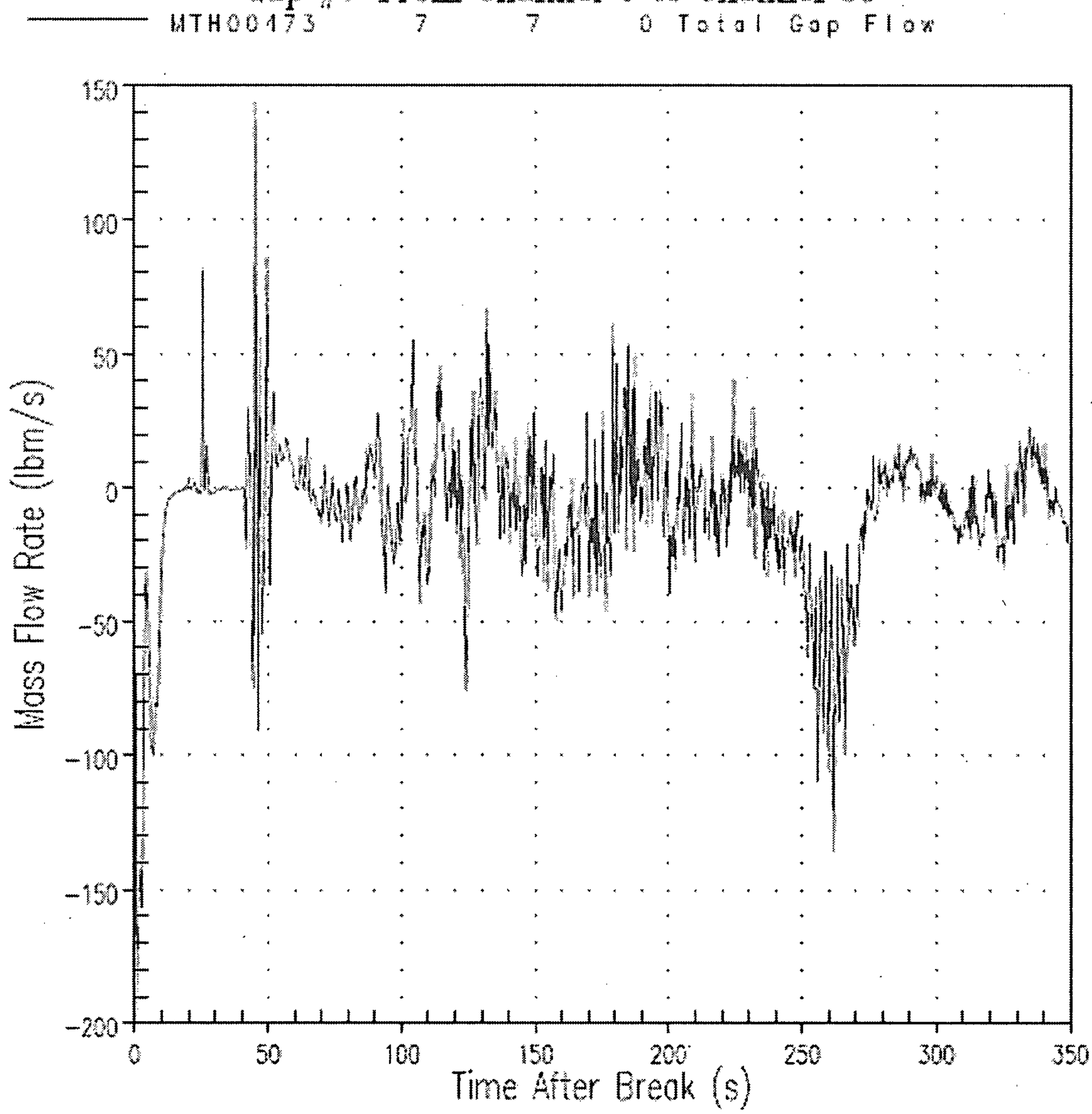
PTN Limiting PCT Case

Figure d27: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Bottom of Section
Gap #72: From Channel 6 to Channel 63



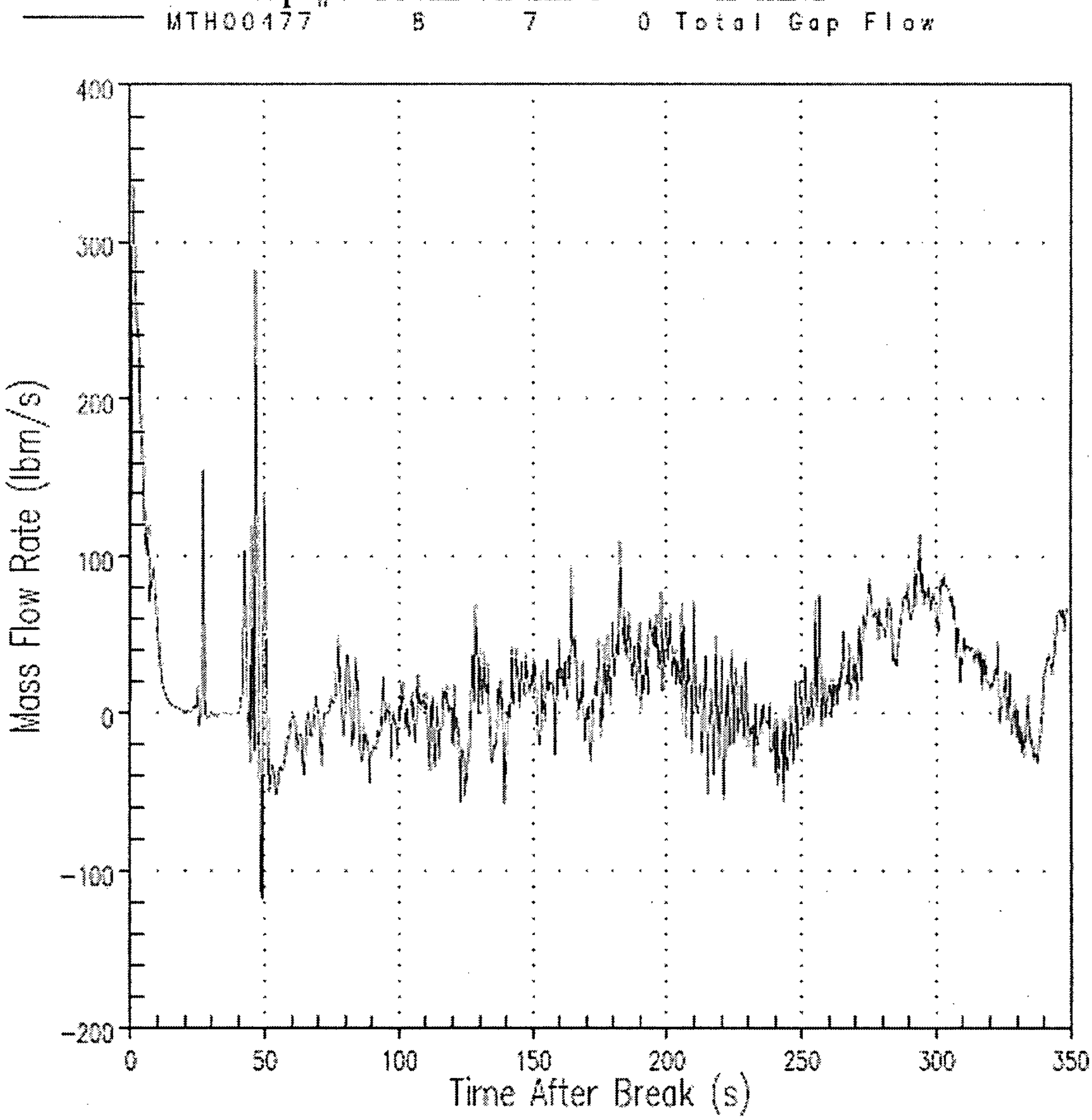
PTN Limiting PCT Case

Figure d28: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Approximately Mid-Section
Gap #7: From Channel 6 to Channel 58



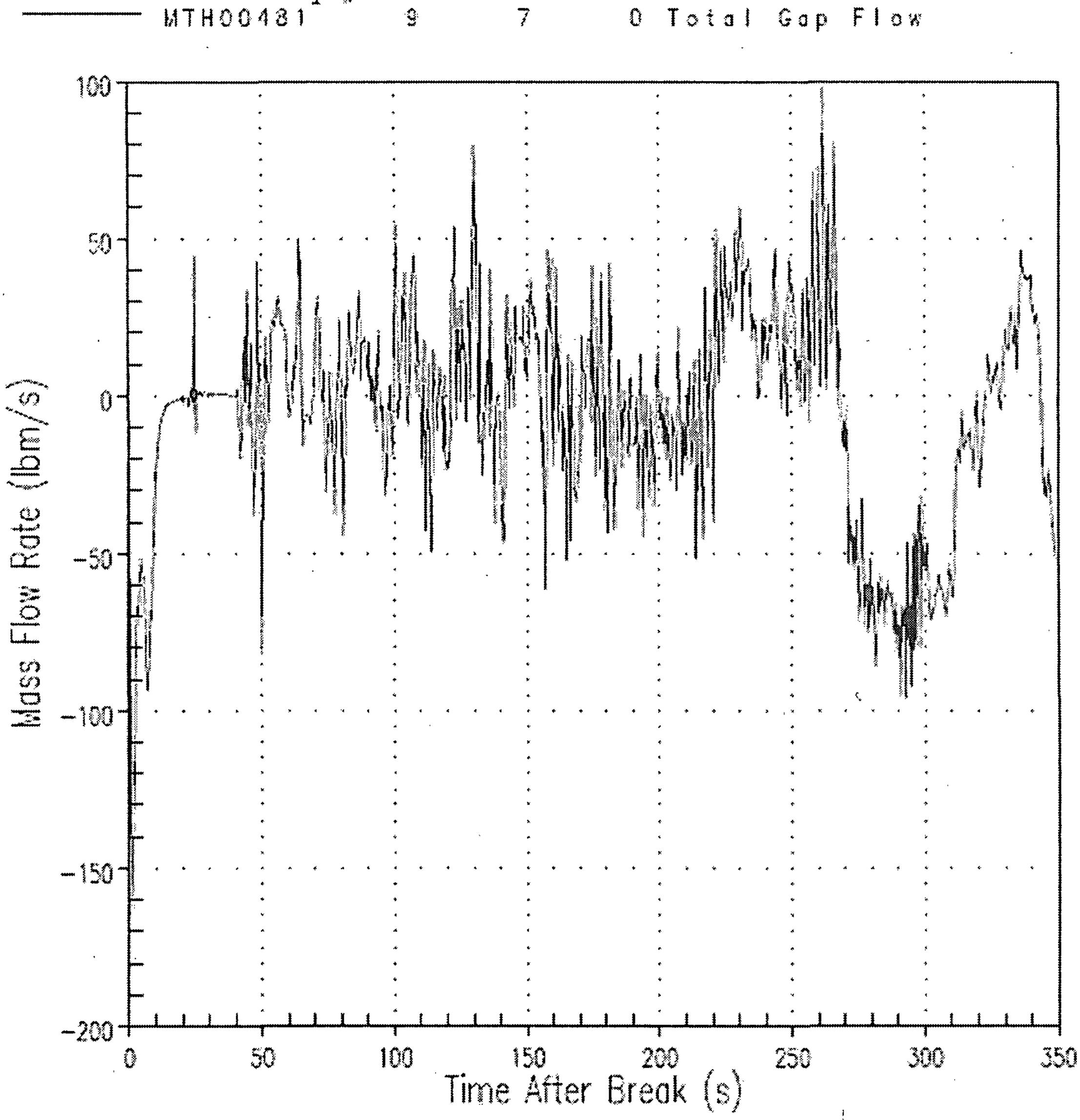
PTN Limiting PCT Case

Figure d29: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Approximately Mid-Section
Gap #8: From Channel 7 to Channel 60



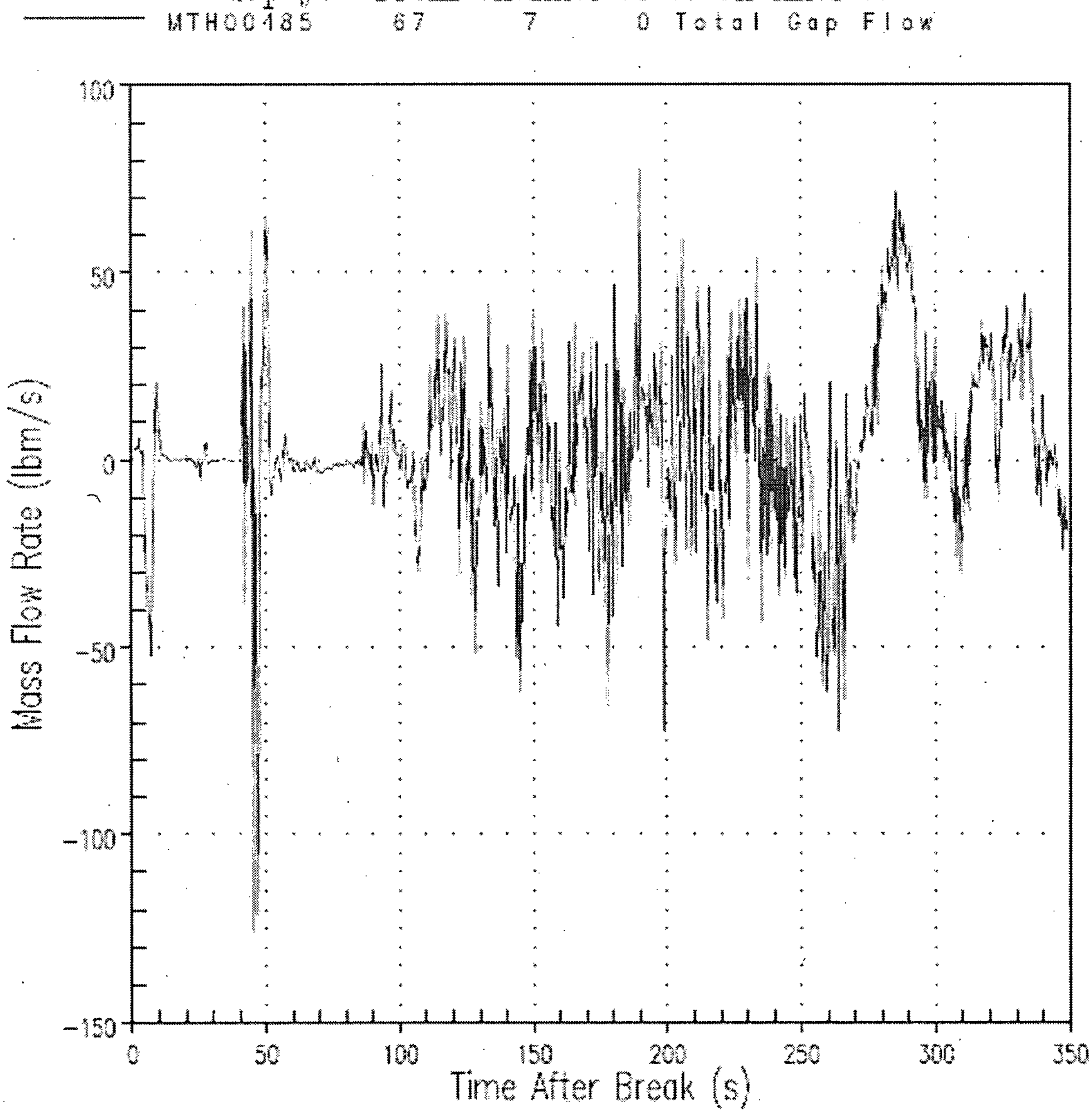
PTN Limiting PCT Case

Figure d30: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Core – Section 3 – at Approximately Mid-Section
Gap #9: From Channel 8 to Channel 62



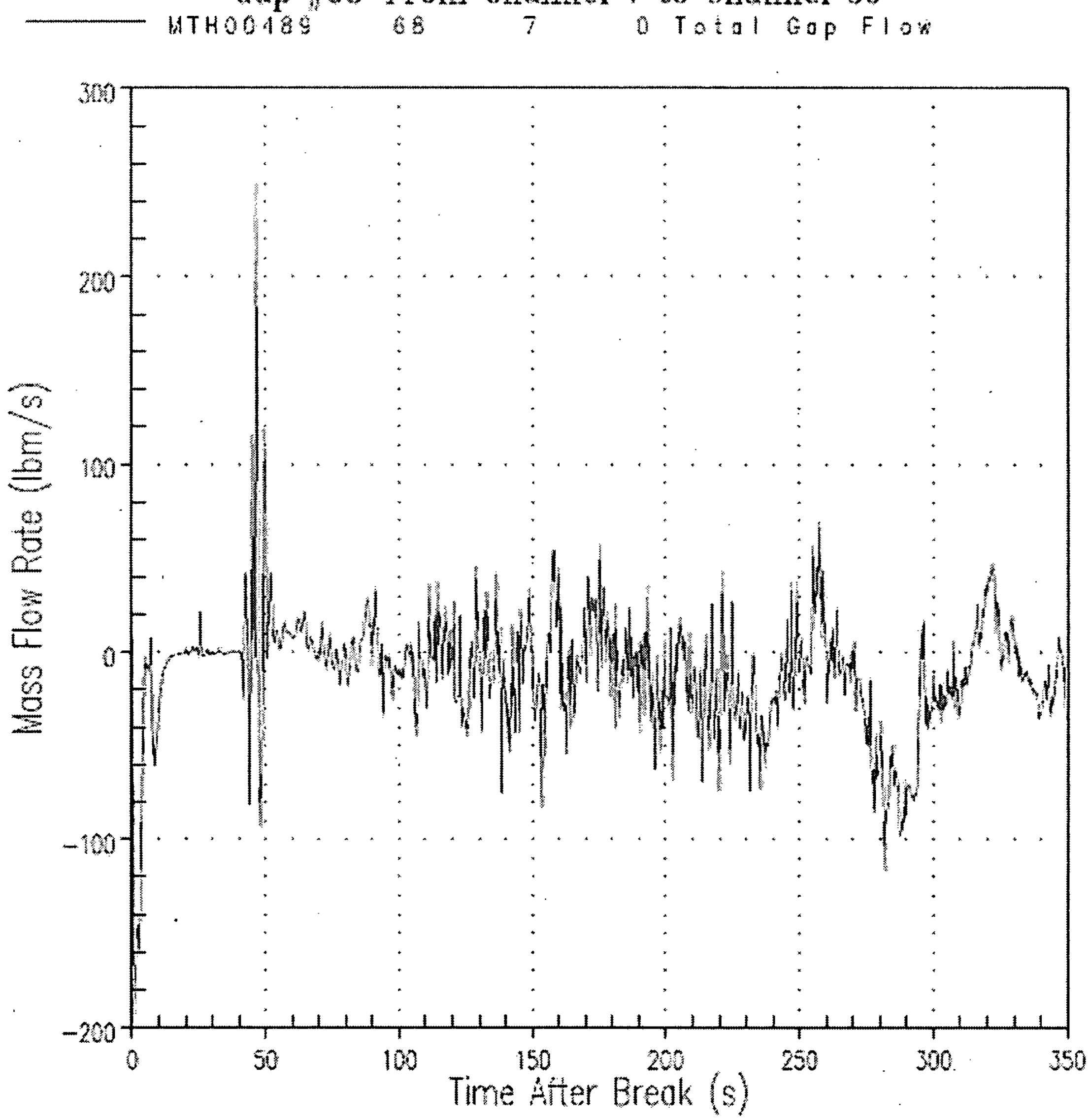
PTN Limiting PCT Case

Figure d31: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Approximately Mid–Section
Gap #67: From Channel 58 to Channel 59



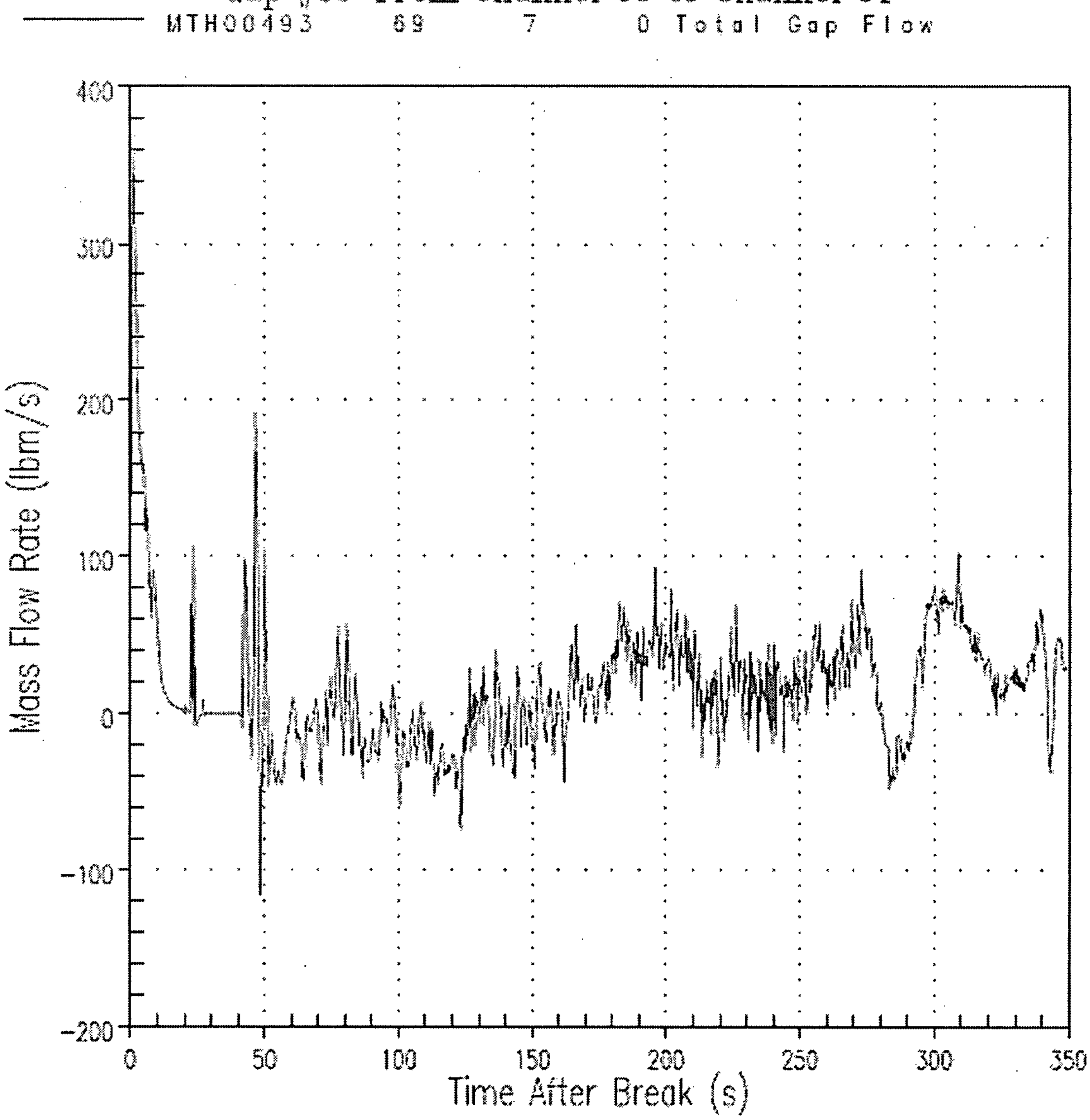
PTN Limiting PCT Case

Figure d32: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Approximately Mid-Section
Gap #68: From Channel 7 to Channel 59



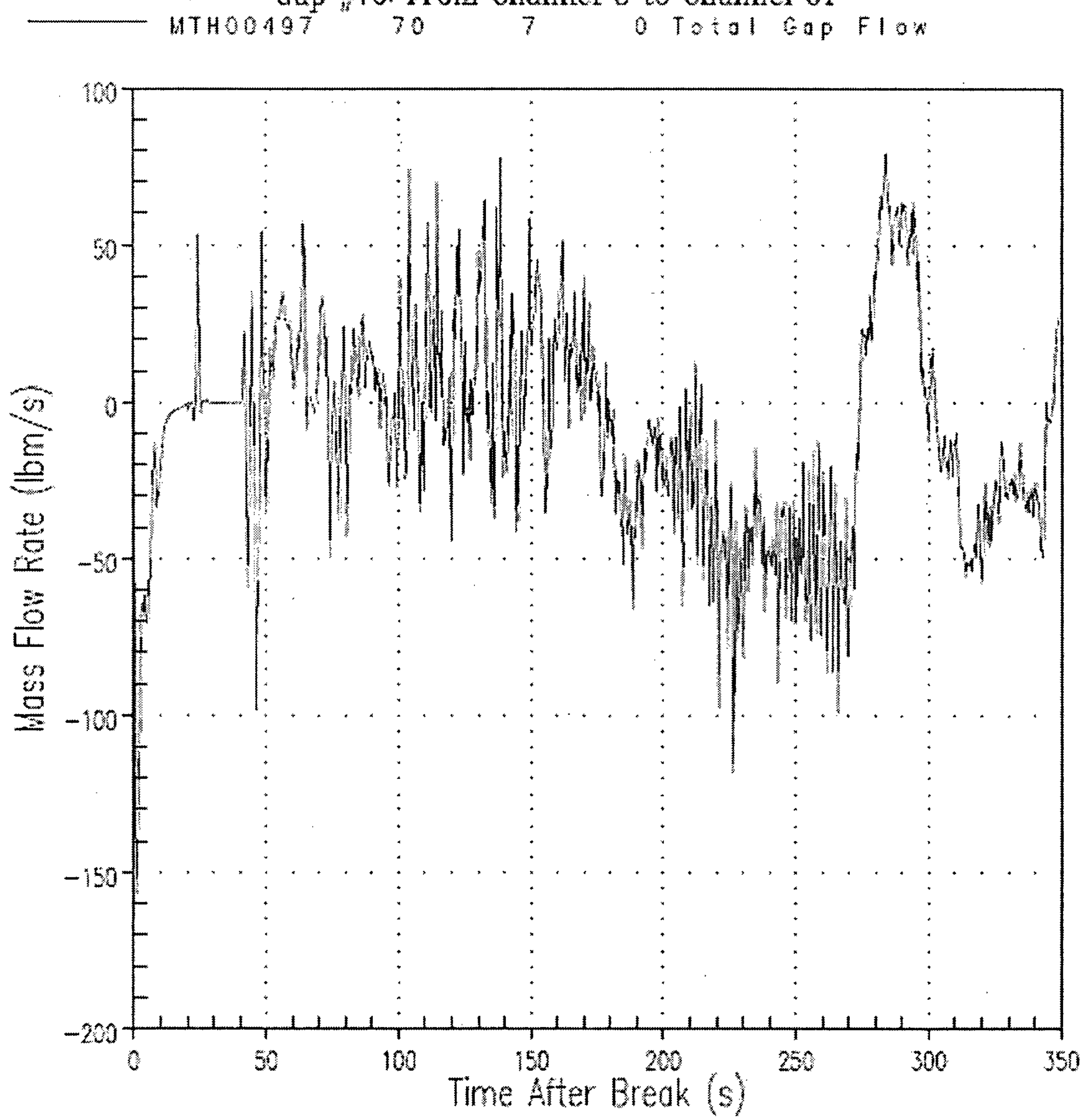
PTN Limiting PCT Case

Figure d33: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Approximately Mid-Section
Gap #69: From Channel 60 to Channel 61



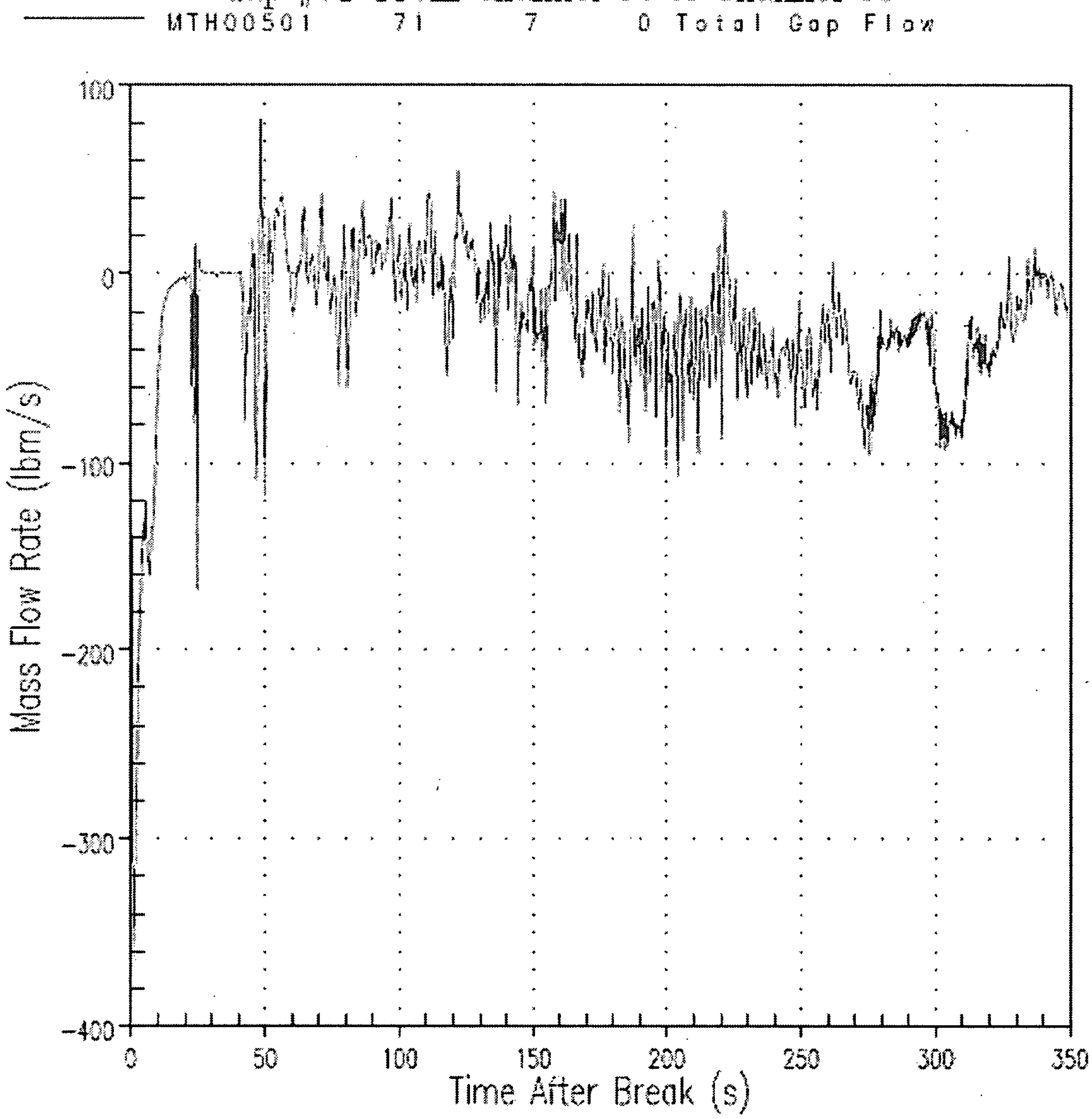
PTN Limiting PCT Case

Figure d34: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Core - Section 3 - at Approximately Mid-Section
Gap #70: From Channel 8 to Channel 61



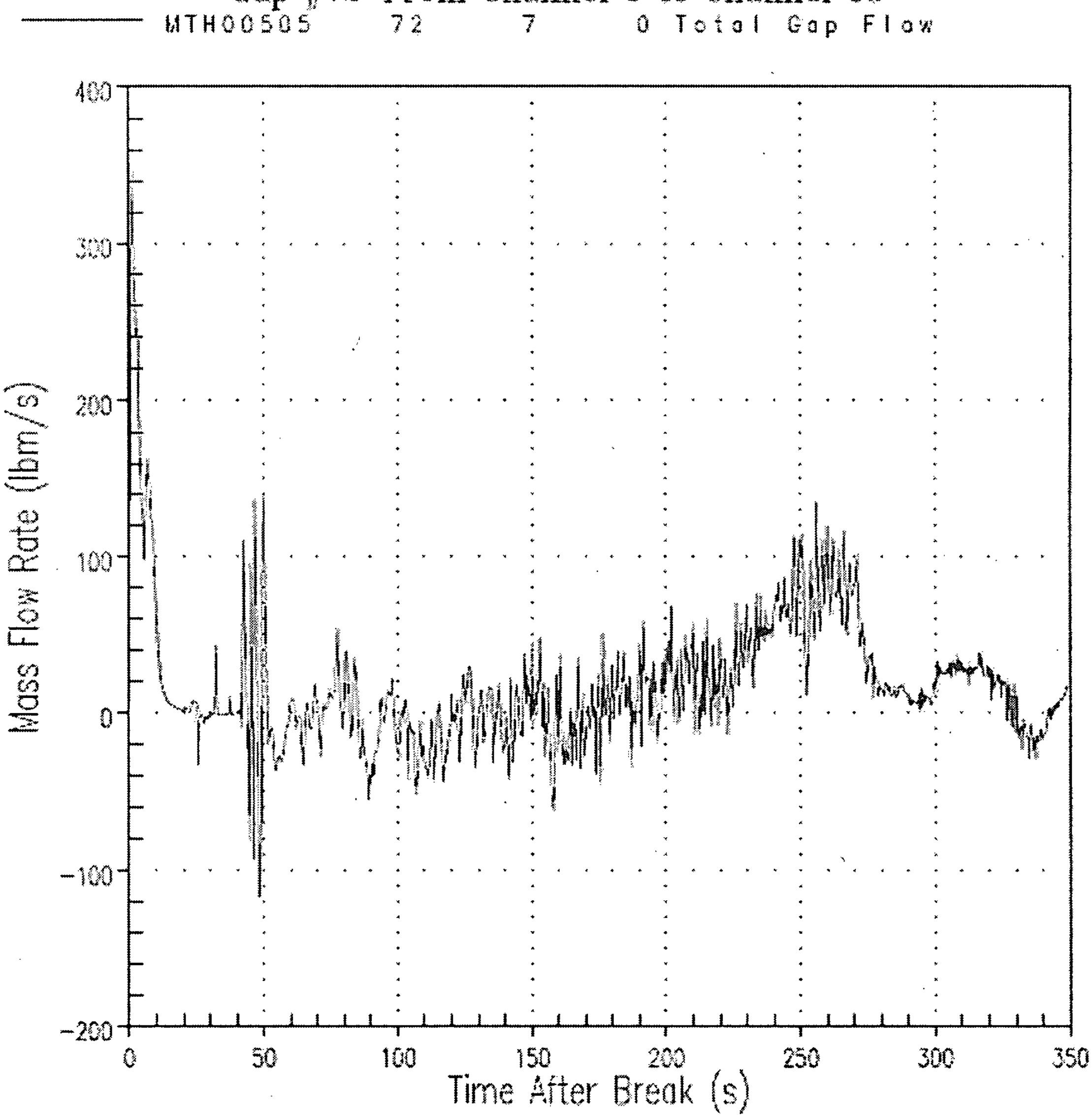
PTN Limiting PCT Case

Figure d35: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Approximately Mid-Section
Gap #71: From Channel 62 to Channel 63



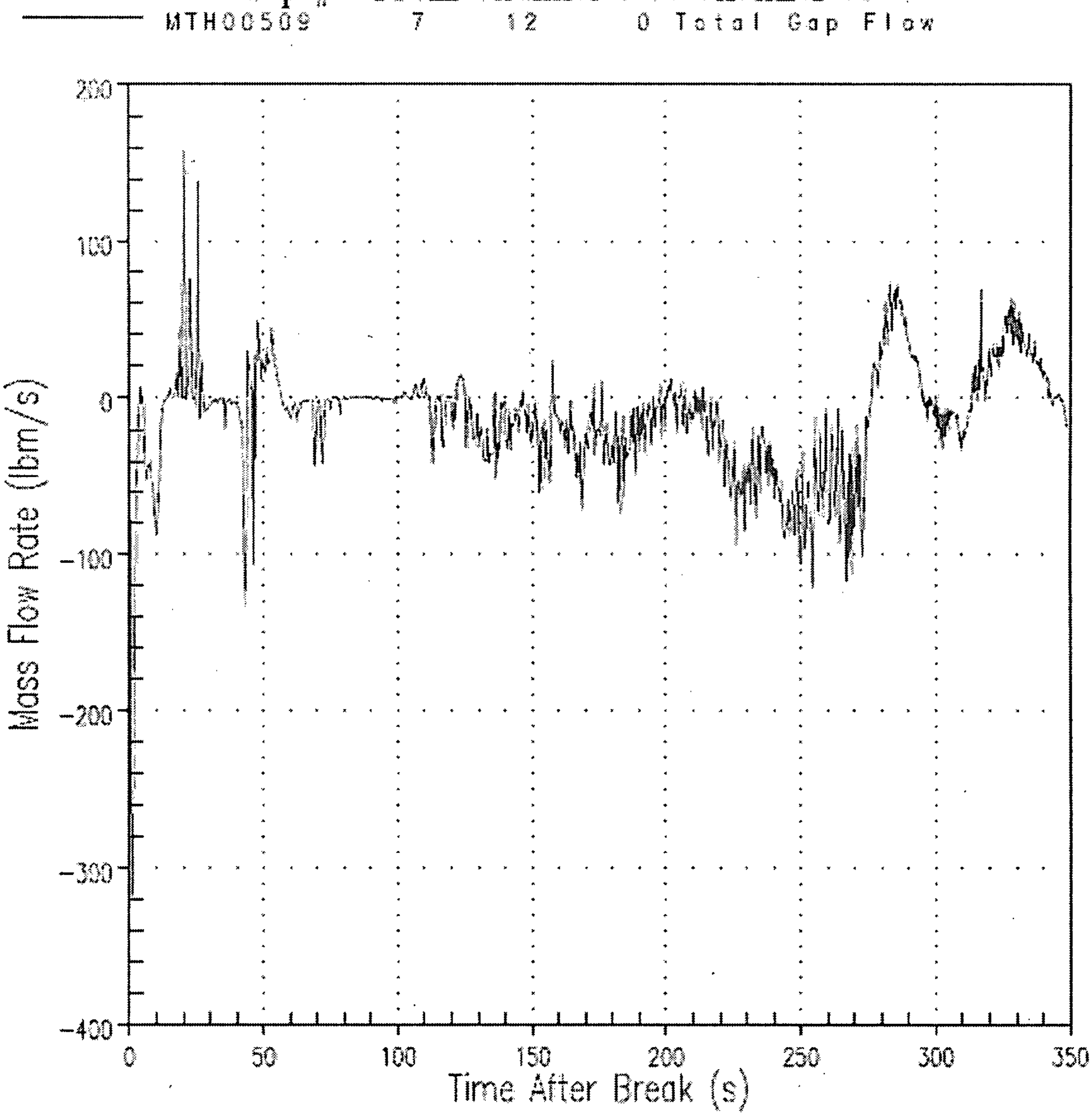
PTN Limiting PCT Case

Figure d36: Total Gap Flow between Intact Loop Downcomer Channels
Core - Section 3 - at Approximately Mid-Section
Gap #72: From Channel 6 to Channel 63



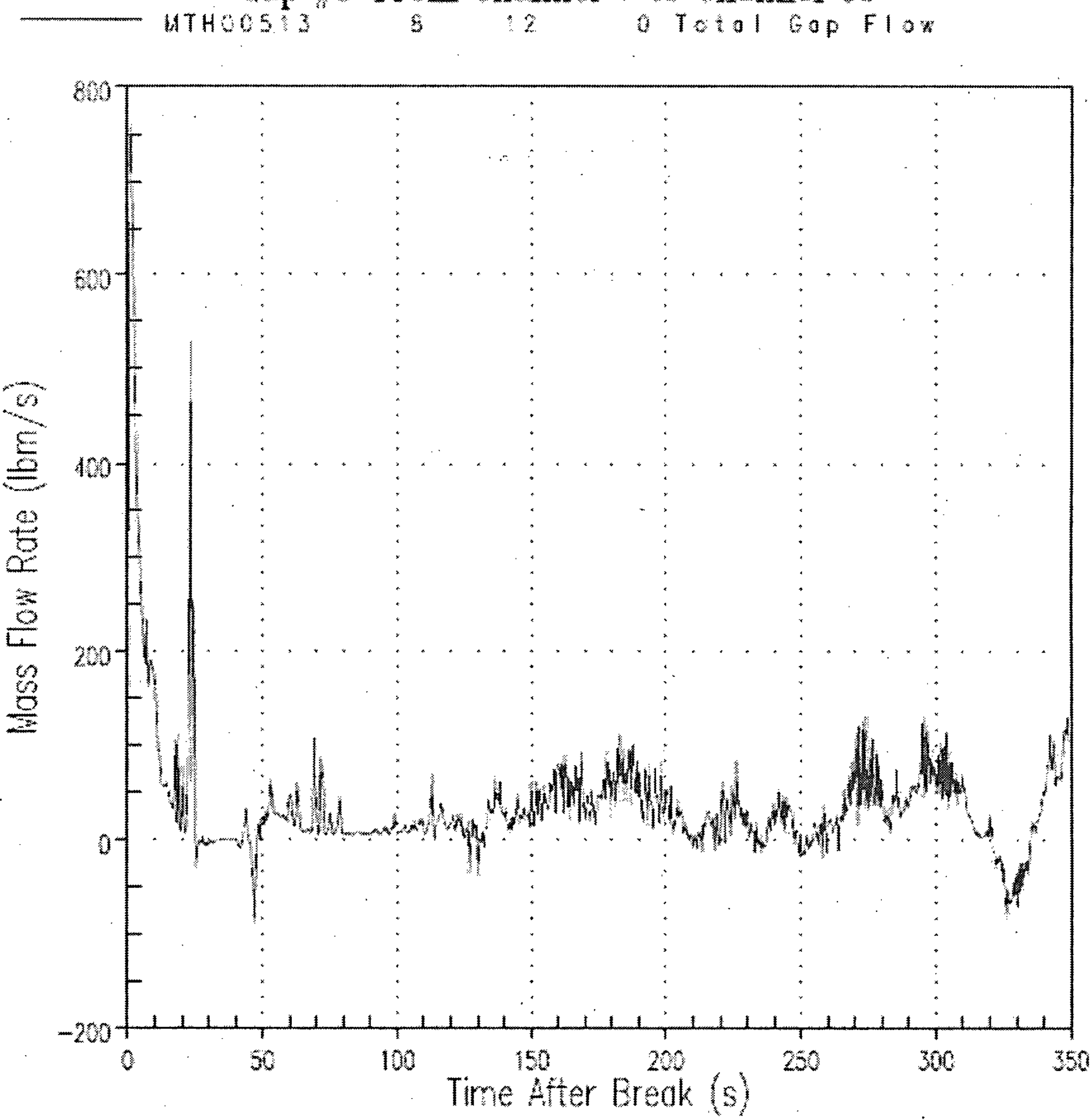
PTN Limiting PCT Case

Figure d37: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Top of Section
Gap #7: From Channel 6 to Channel 58



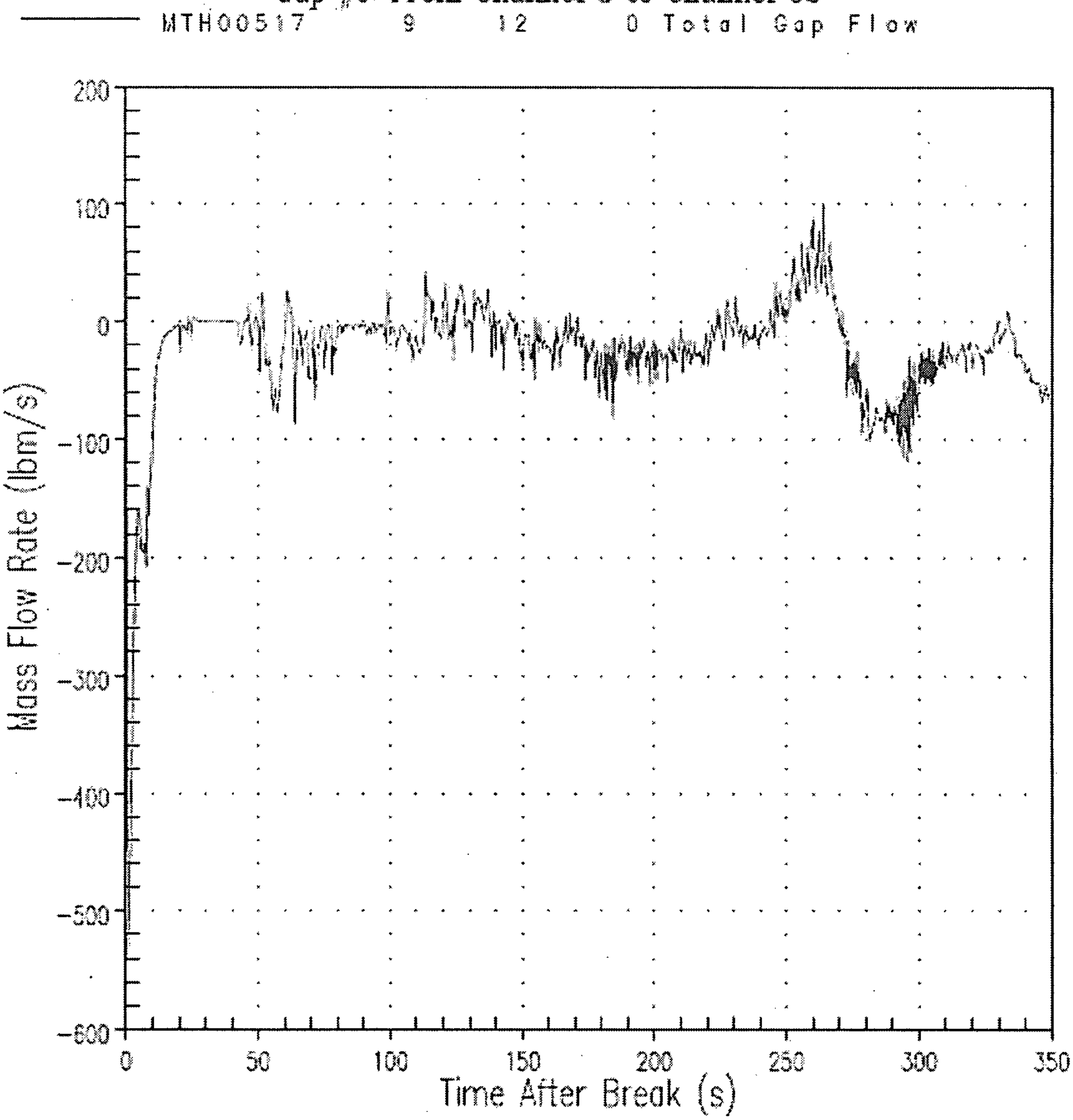
PTN Limiting PCT Case

Figure d38: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Top of Section
Gap #8: From Channel 7 to Channel 60



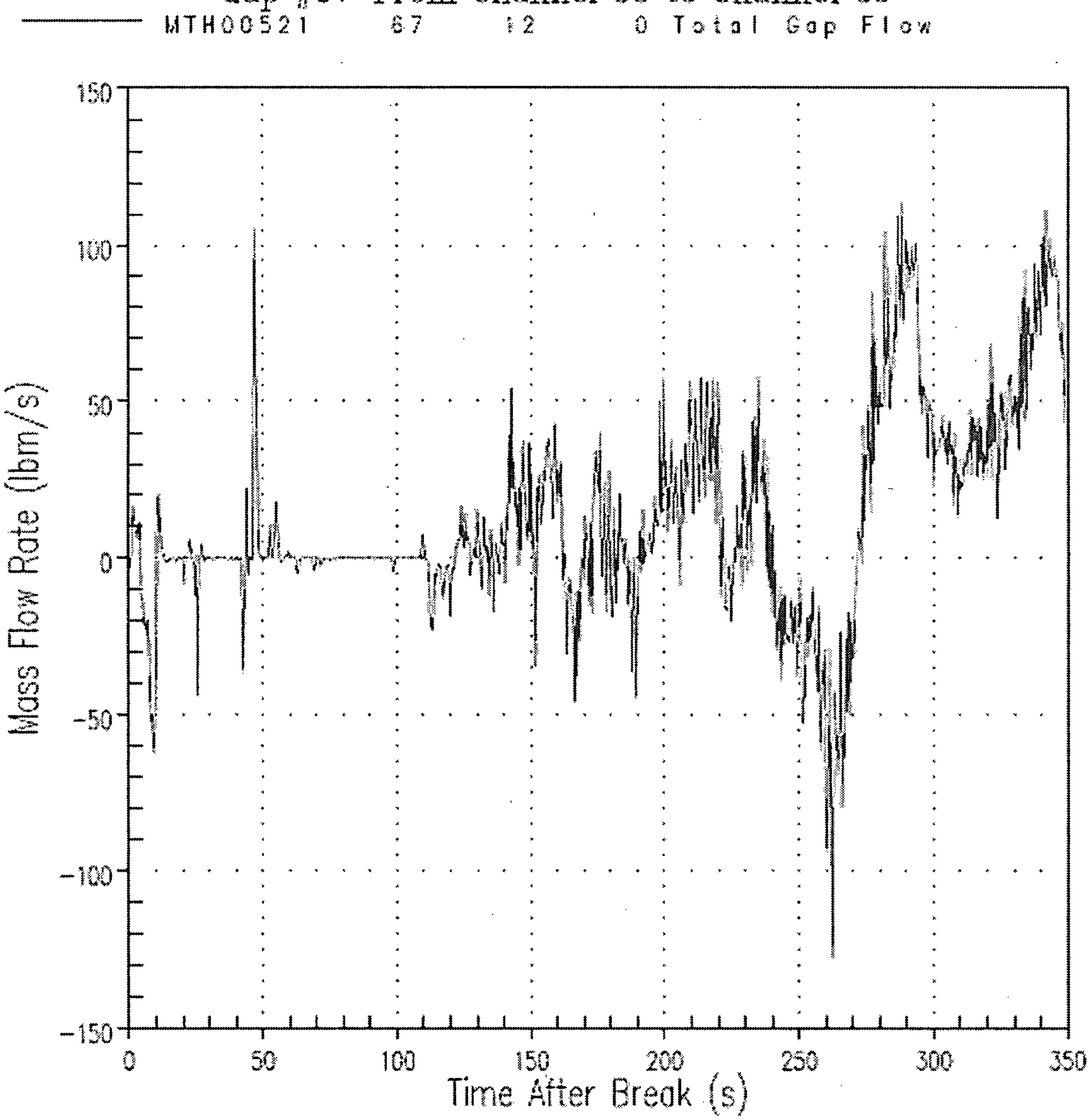
PTN Limiting PCT Case

Figure d39: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Core – Section 3 – at Top of Section
Gap #9: From Channel 8 to Channel 62



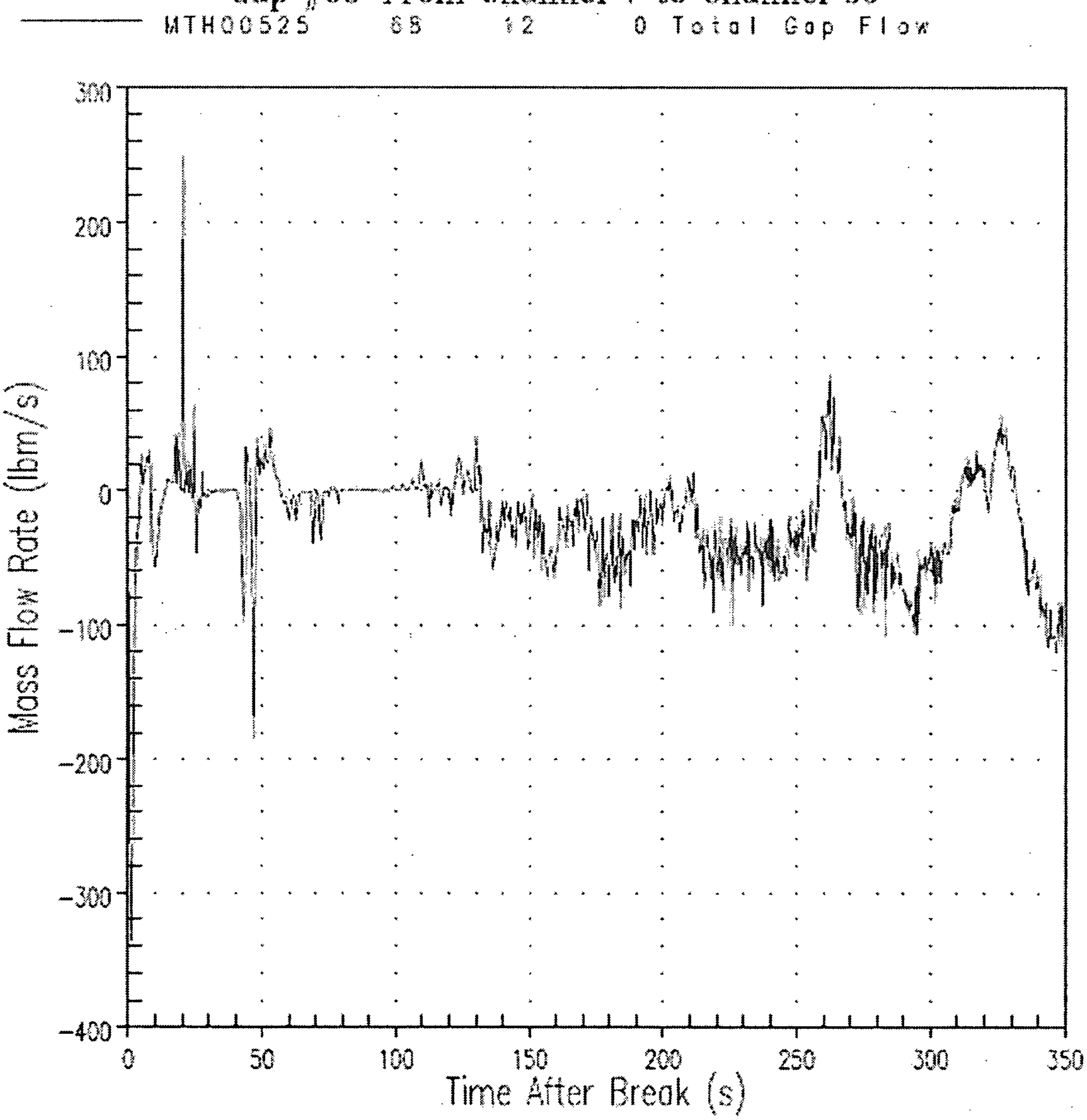
PTN Limiting PCT Case

Figure d40: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Top of Section
Gap #67: From Channel 58 to Channel 59



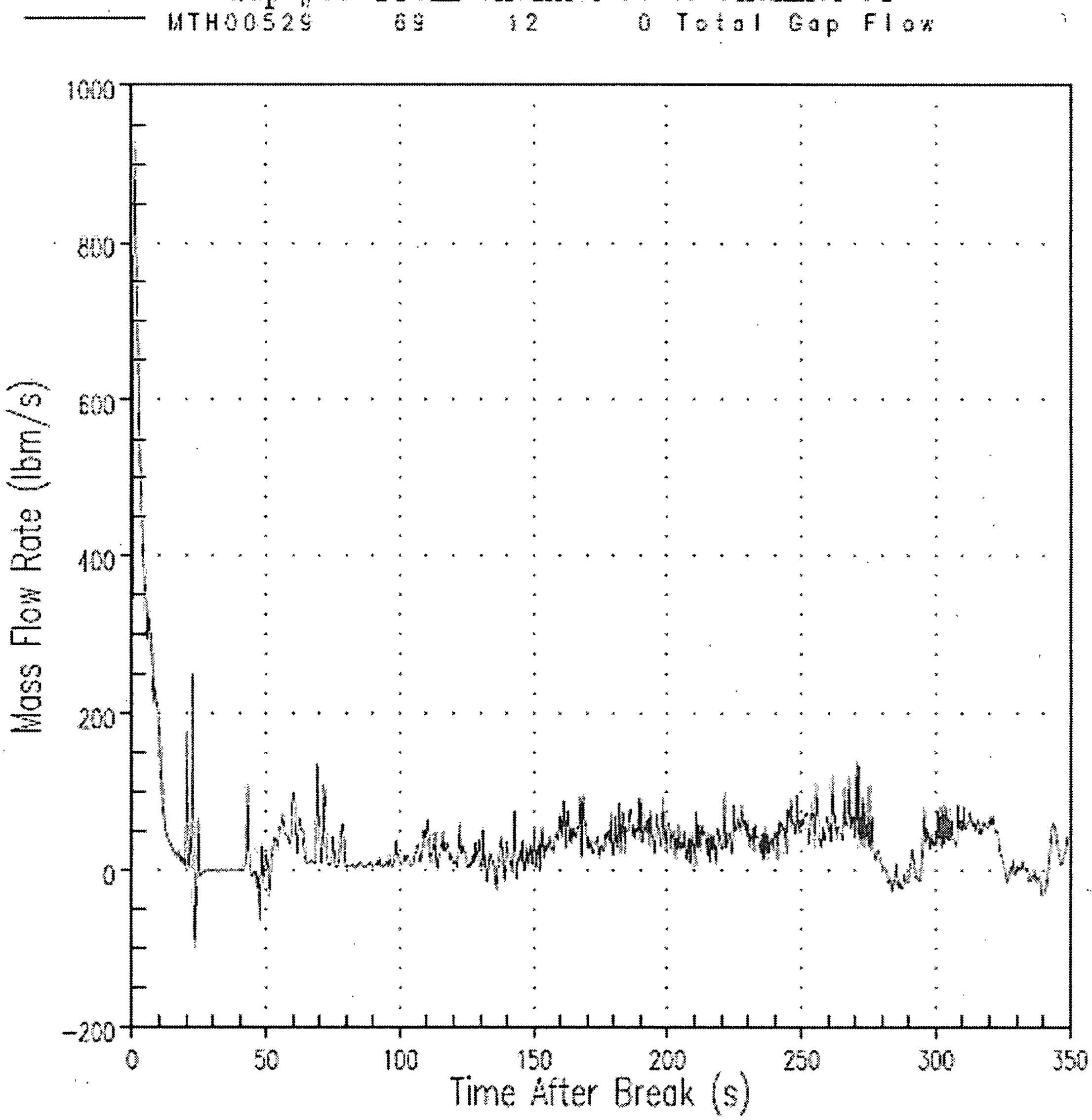
PTN Limiting PCT Case

Figure d41: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Top of Section
Gap #68: From Channel 7 to Channel 59



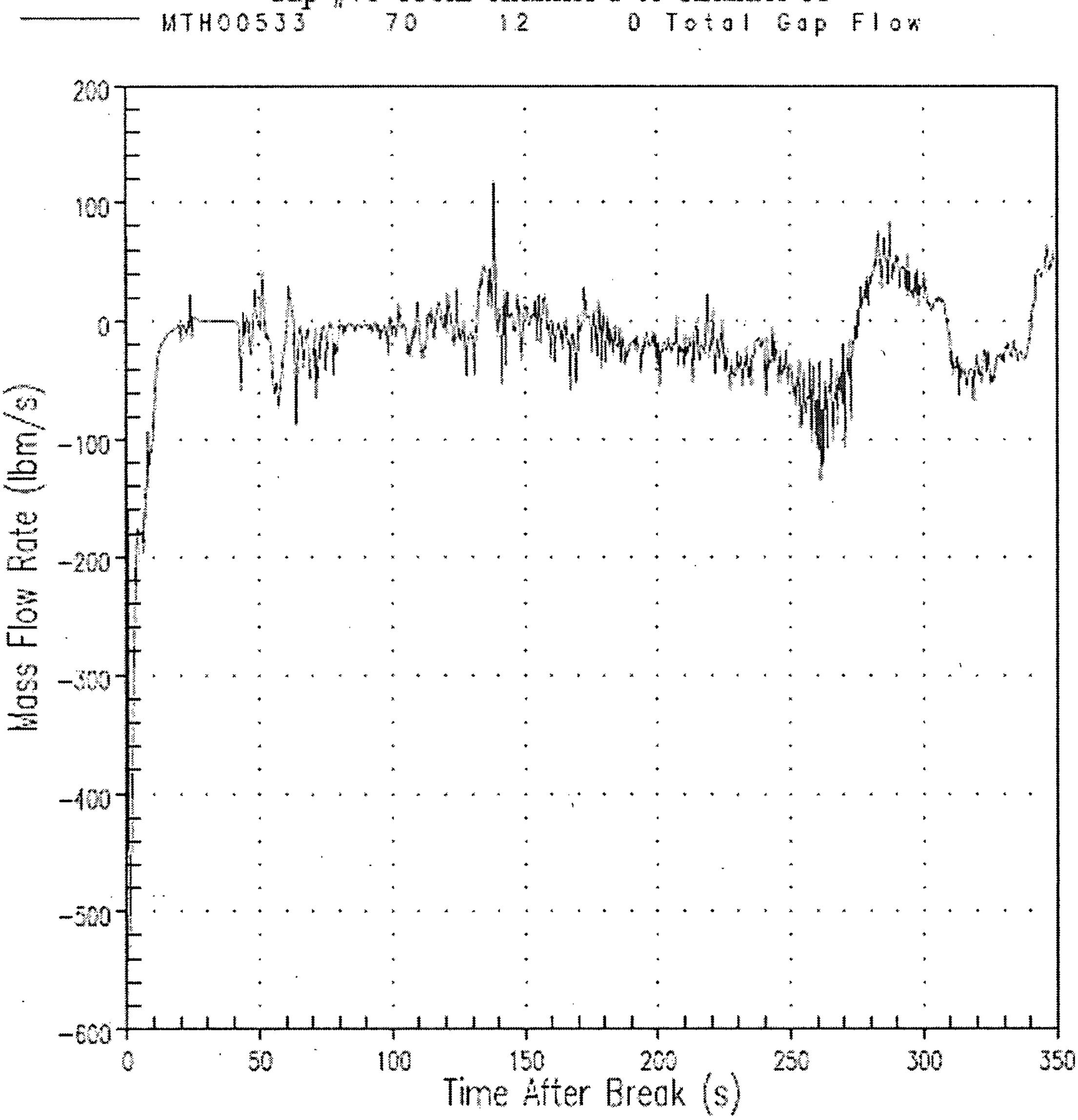
PTN Limiting PCT Case

Figure d42: Total Gap Flow between Intact Loop Downcomer Channels
Core - Section 3 - at Top of Section
Gap #69: From Channel 60 to Channel 61



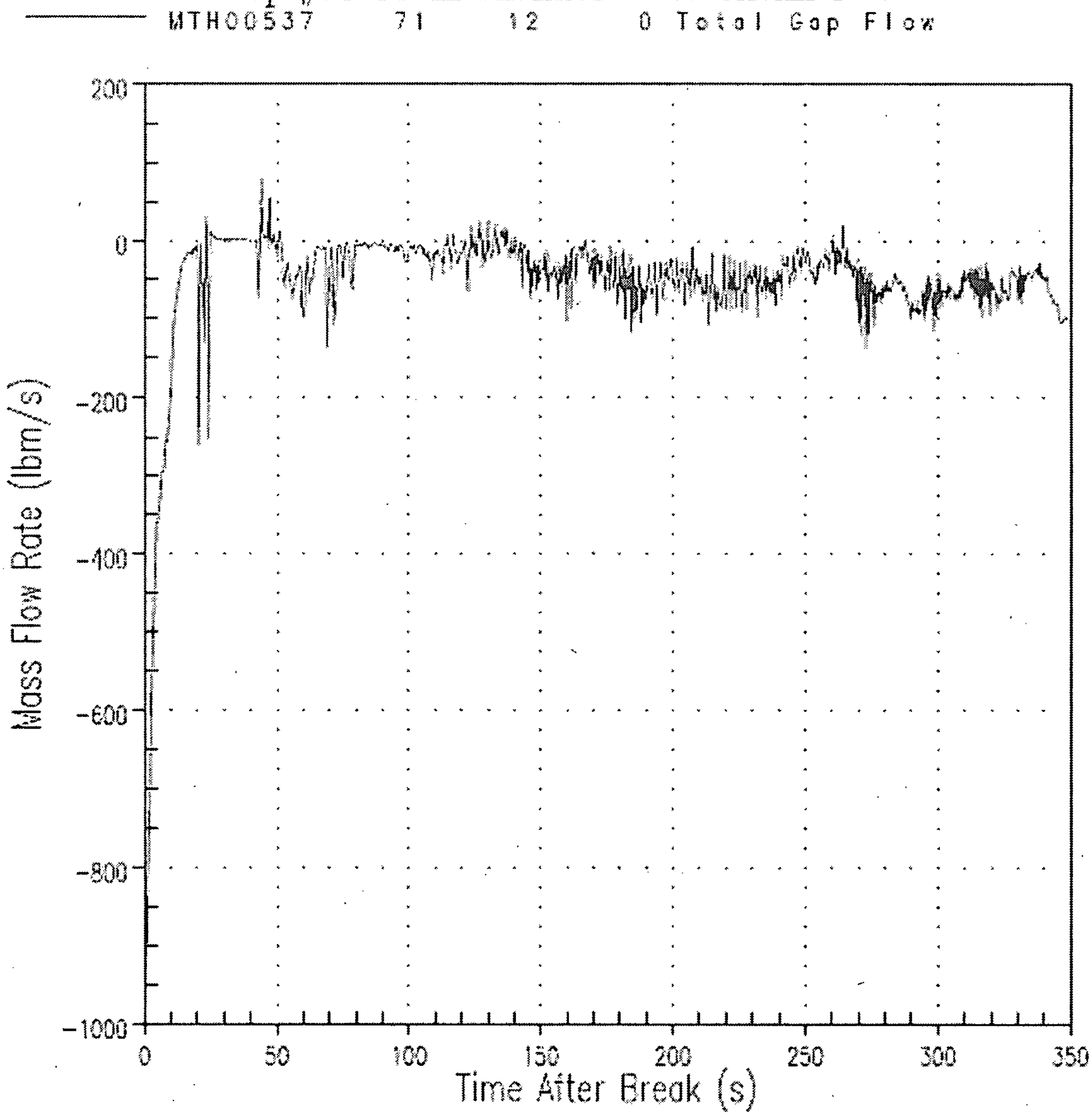
PTN Limiting PCT Case

Figure d43: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Core – Section 3 – at Top of Section
Gap #70: From Channel 8 to Channel 61



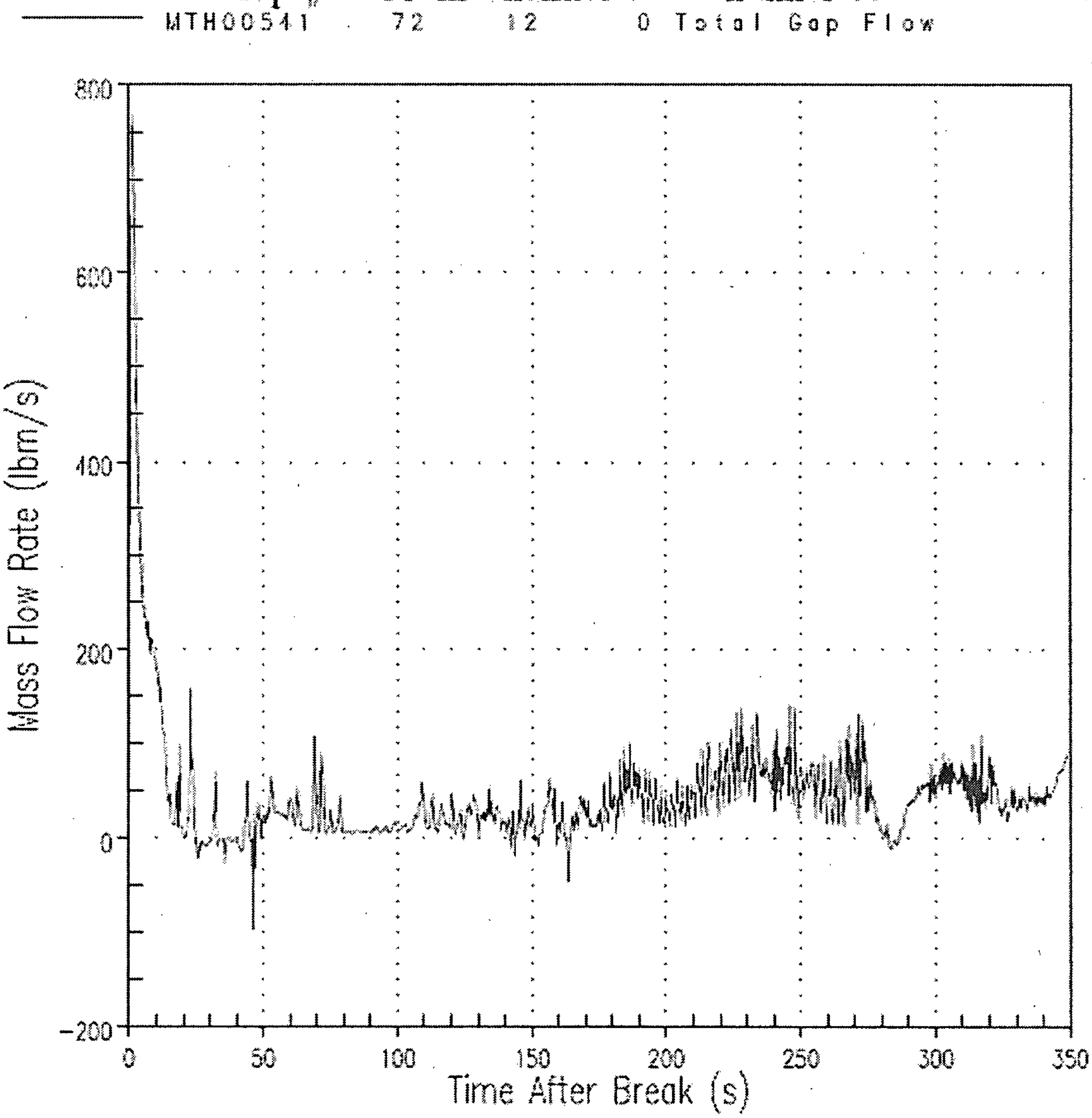
PTN Limiting PCT Case

Figure d44: Total Gap Flow between Intact Loop Downcomer Channels
Core - Section 3 - at Top of Section
Gap #71: From Channel 62 to Channel 63



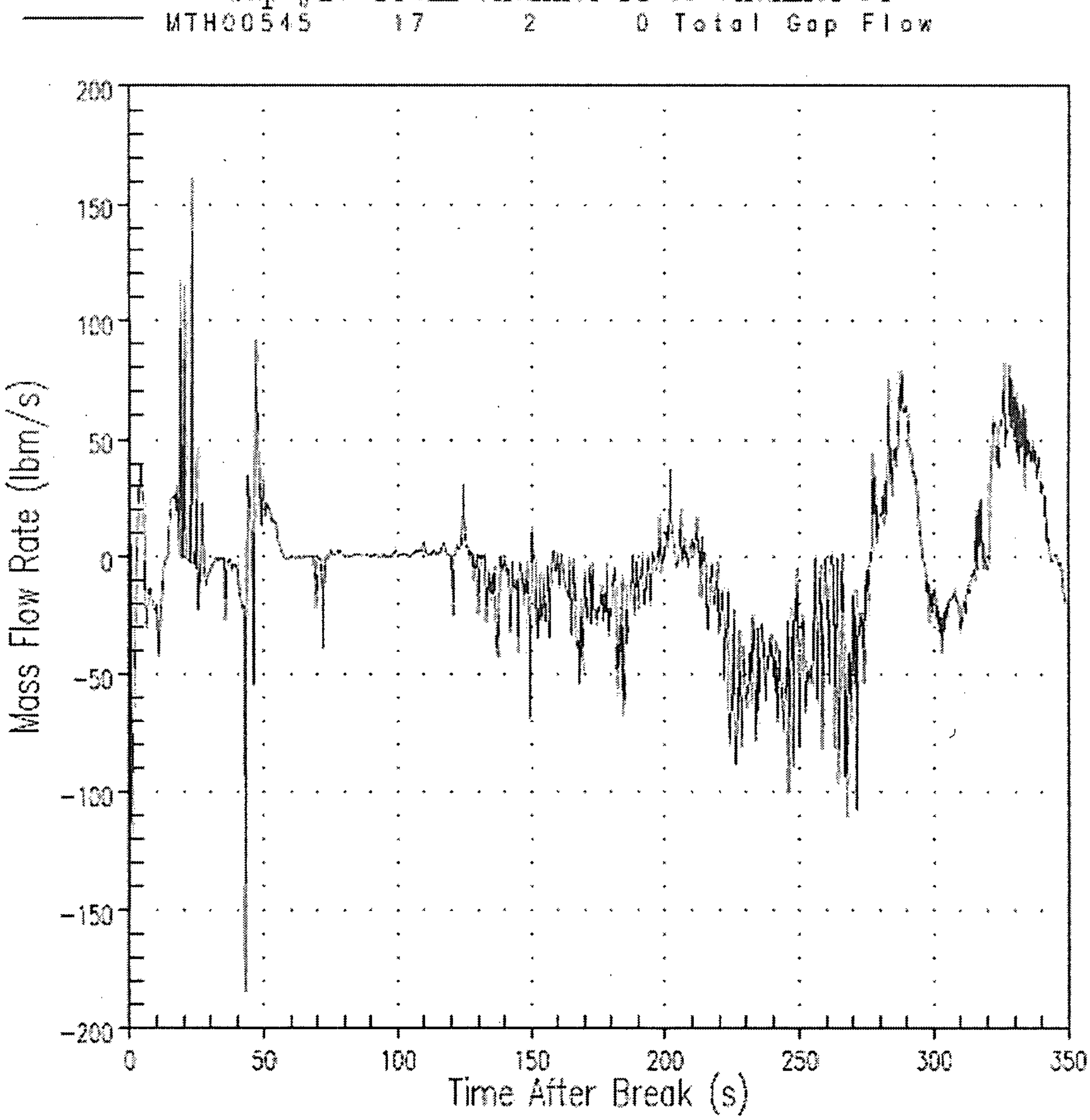
PTN Limiting PCT Case

Figure d45: Total Gap Flow between Intact Loop Downcomer Channels
Core – Section 3 – at Top of Section
Gap #72: From Channel 6 to Channel 63



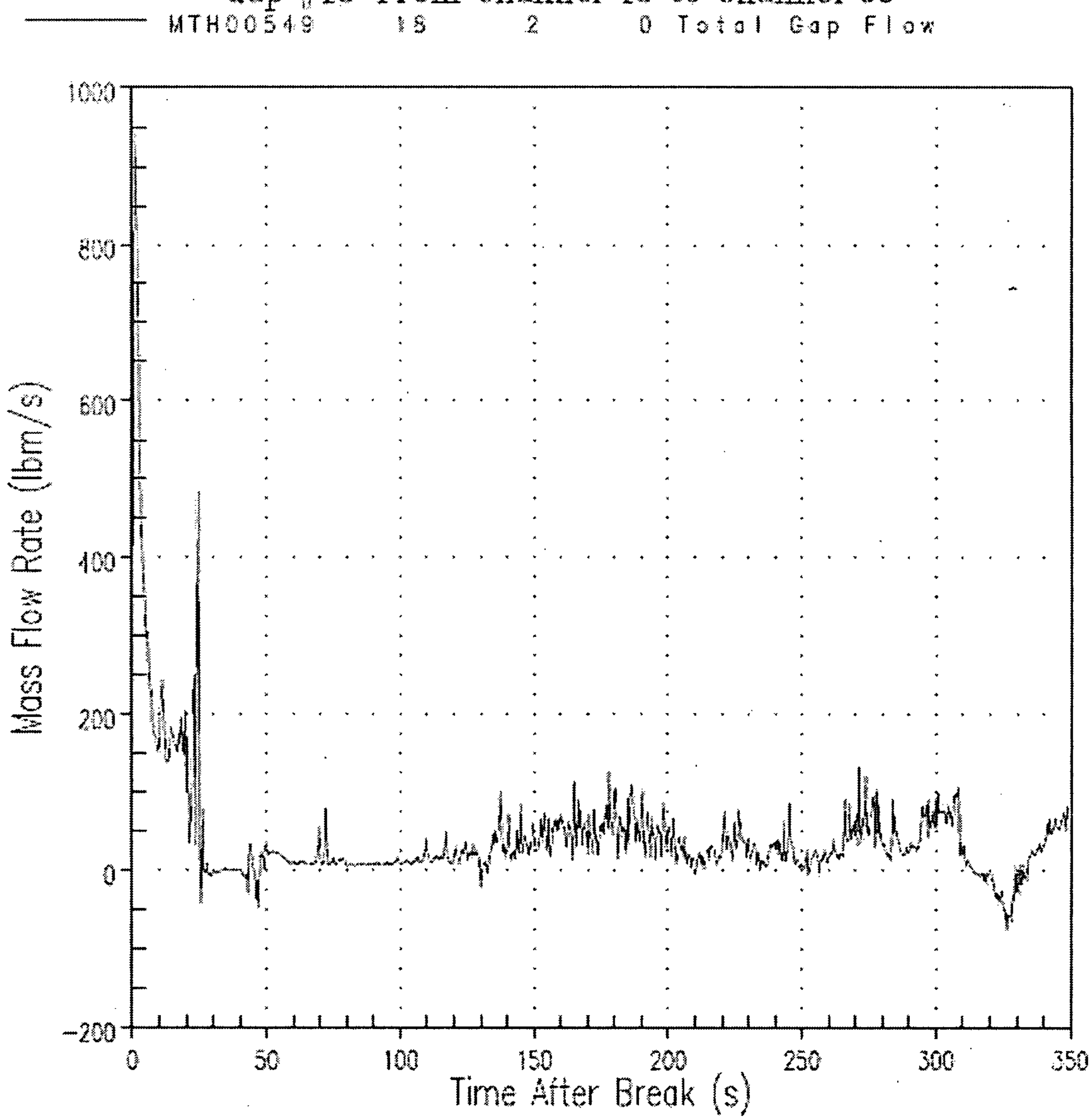
PTN Limiting PCT Case

Figure d46: Total Gap Flow between Intact Loop Downcomer Channels
CCFL – Section 4
Gap #17: From Channel 14 to Channel 64



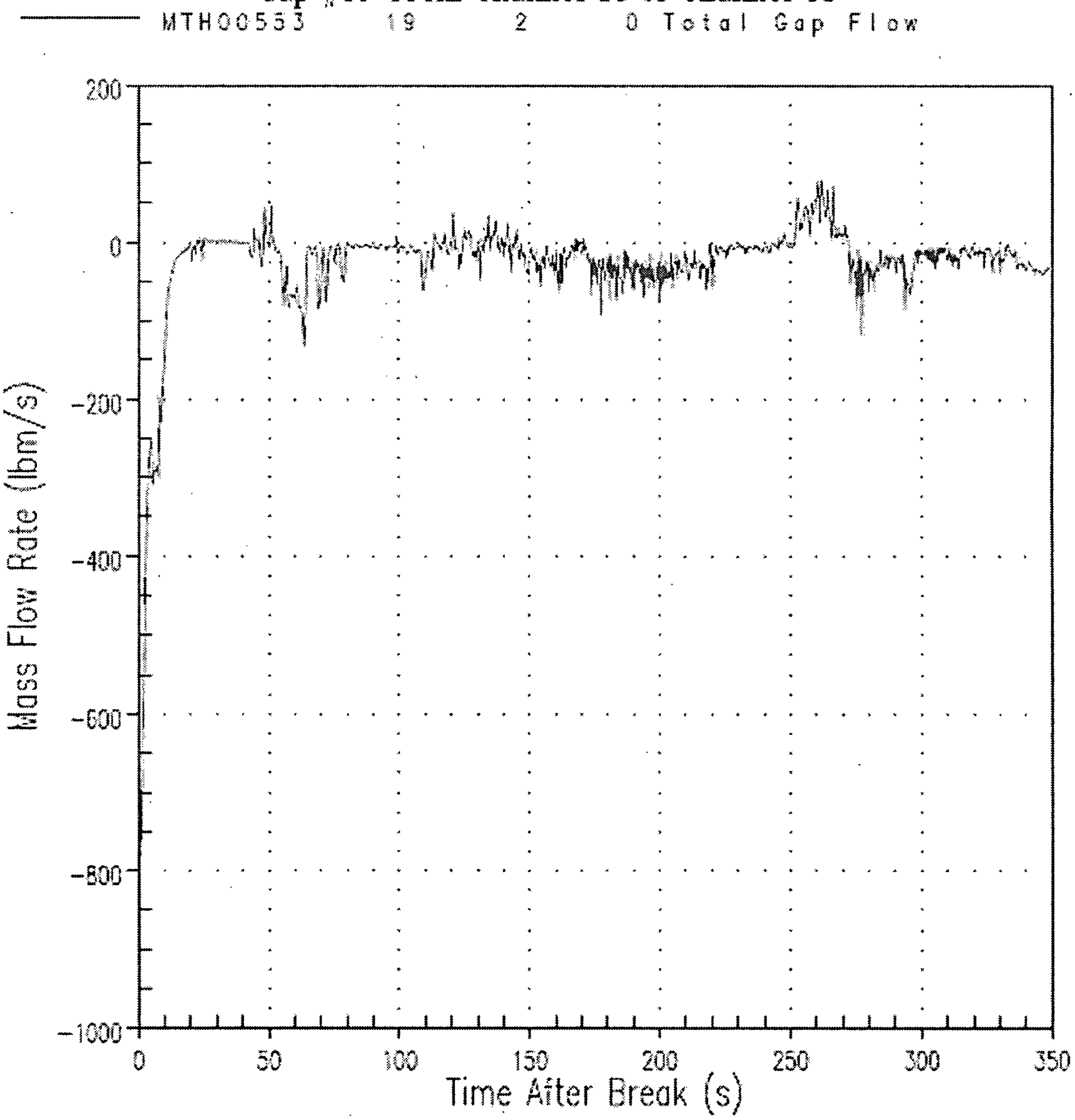
PTN Limiting PCT Case

Figure d47: Total Gap Flow between Intact Loop Downcomer Channels
CCFL – Section 4
Gap #18: From Channel 15 to Channel 66



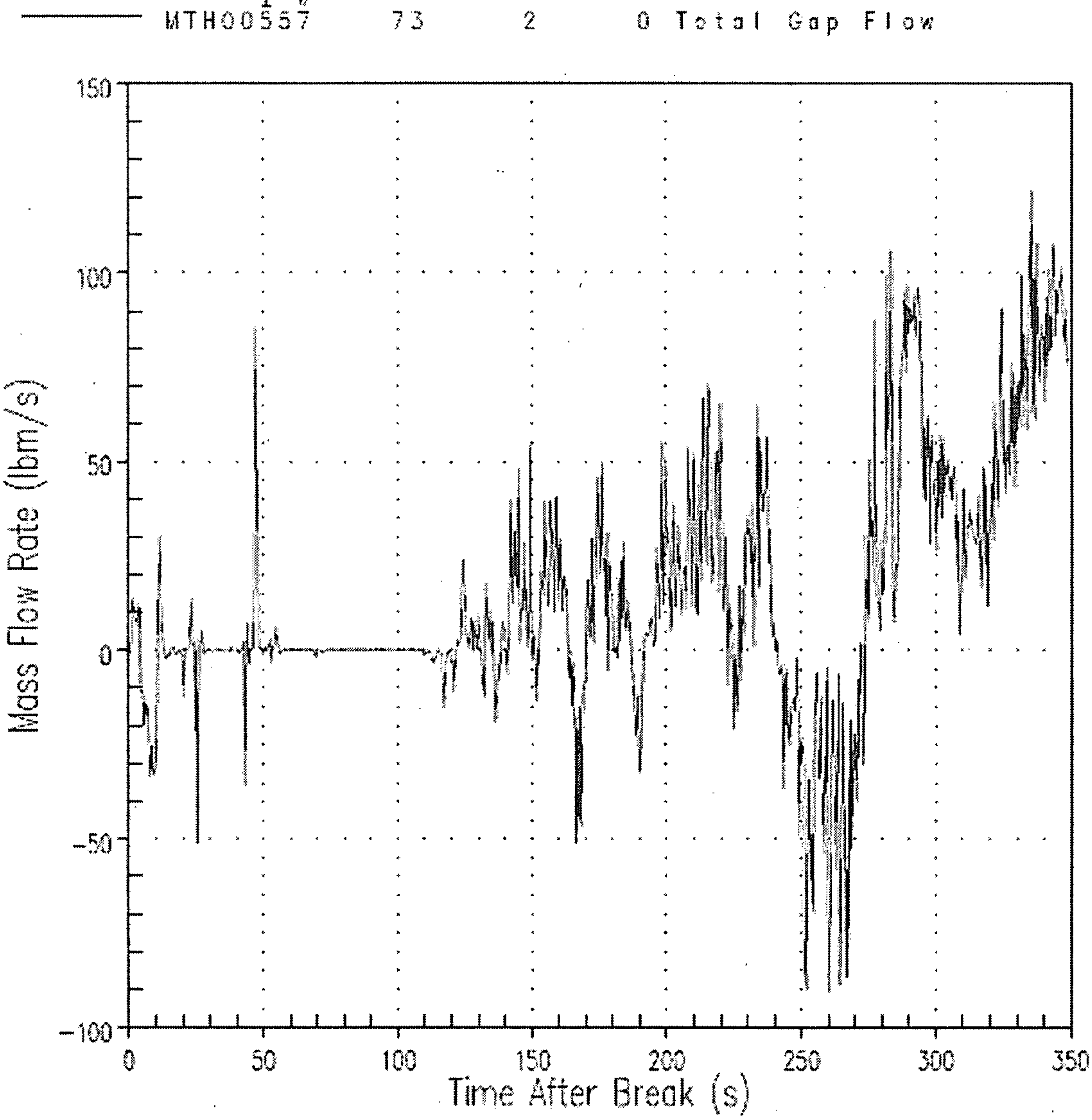
PTN Limiting PCT Case

Figure d48: Total Gap Flow between Intact & Broken Loop Downcomer Channels
CCFL – Section 4
Gap #19: From Channel 16 to Channel 68



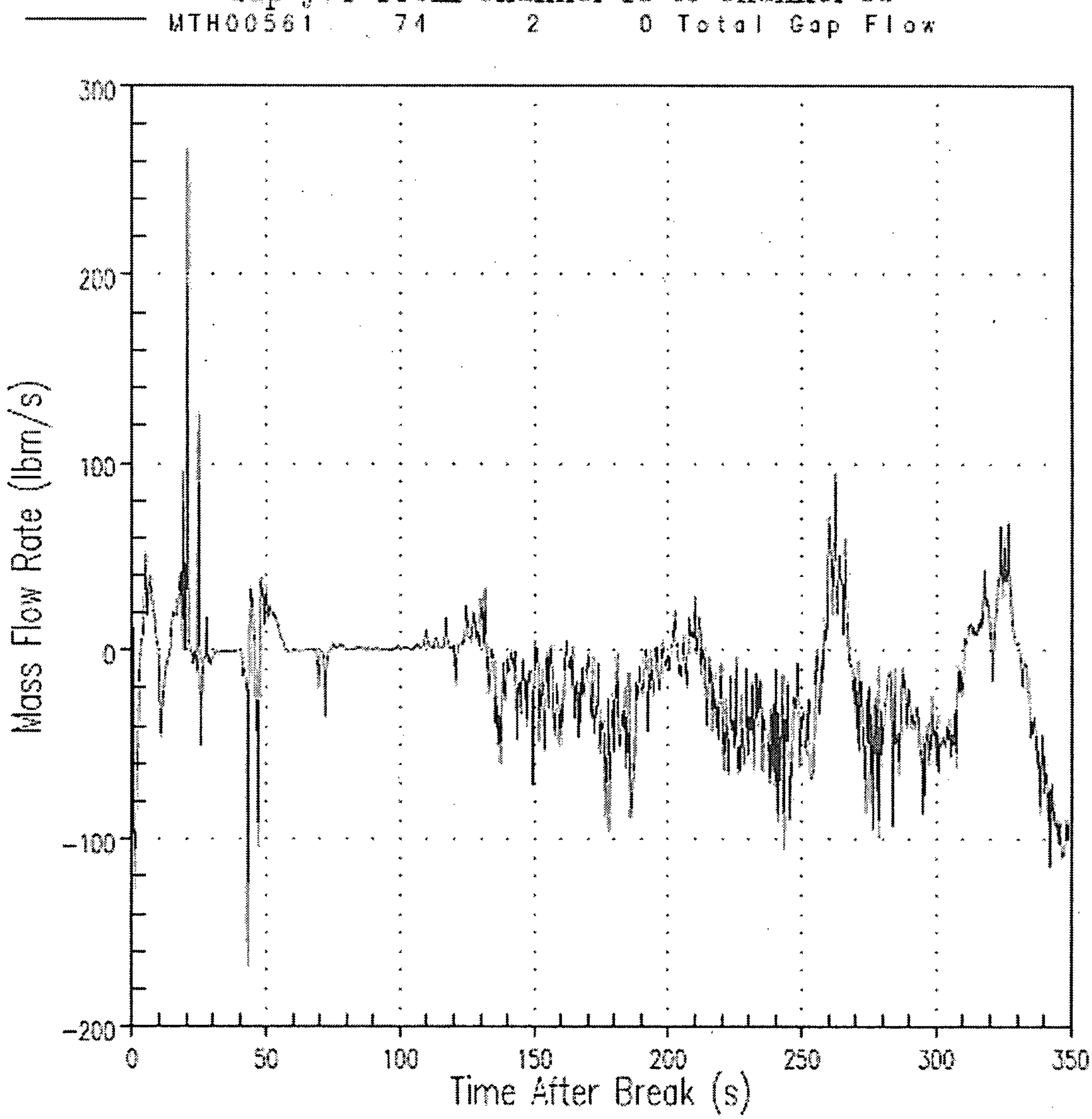
PTN Limiting PCT Case

Figure d49: Total Gap Flow between Intact Loop Downcomer Channels
CCFL – Section 4
Gap #73: From Channel 64 to Channel 65



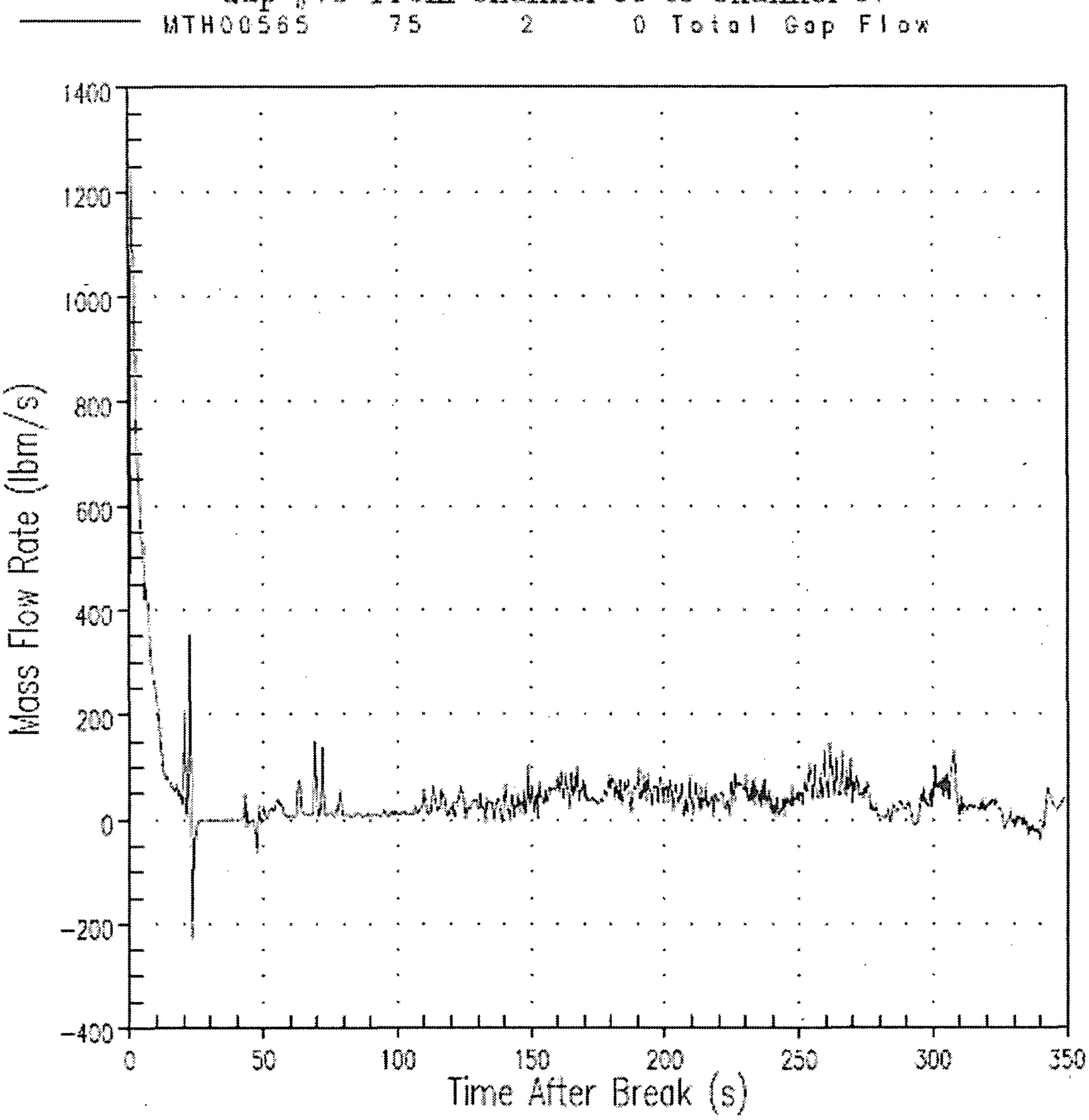
PTN Limiting PCT Case

Figure d50: Total Gap Flow between Intact Loop Downcomer Channels
CCFL – Section 4
Gap #74: From Channel 15 to Channel 65



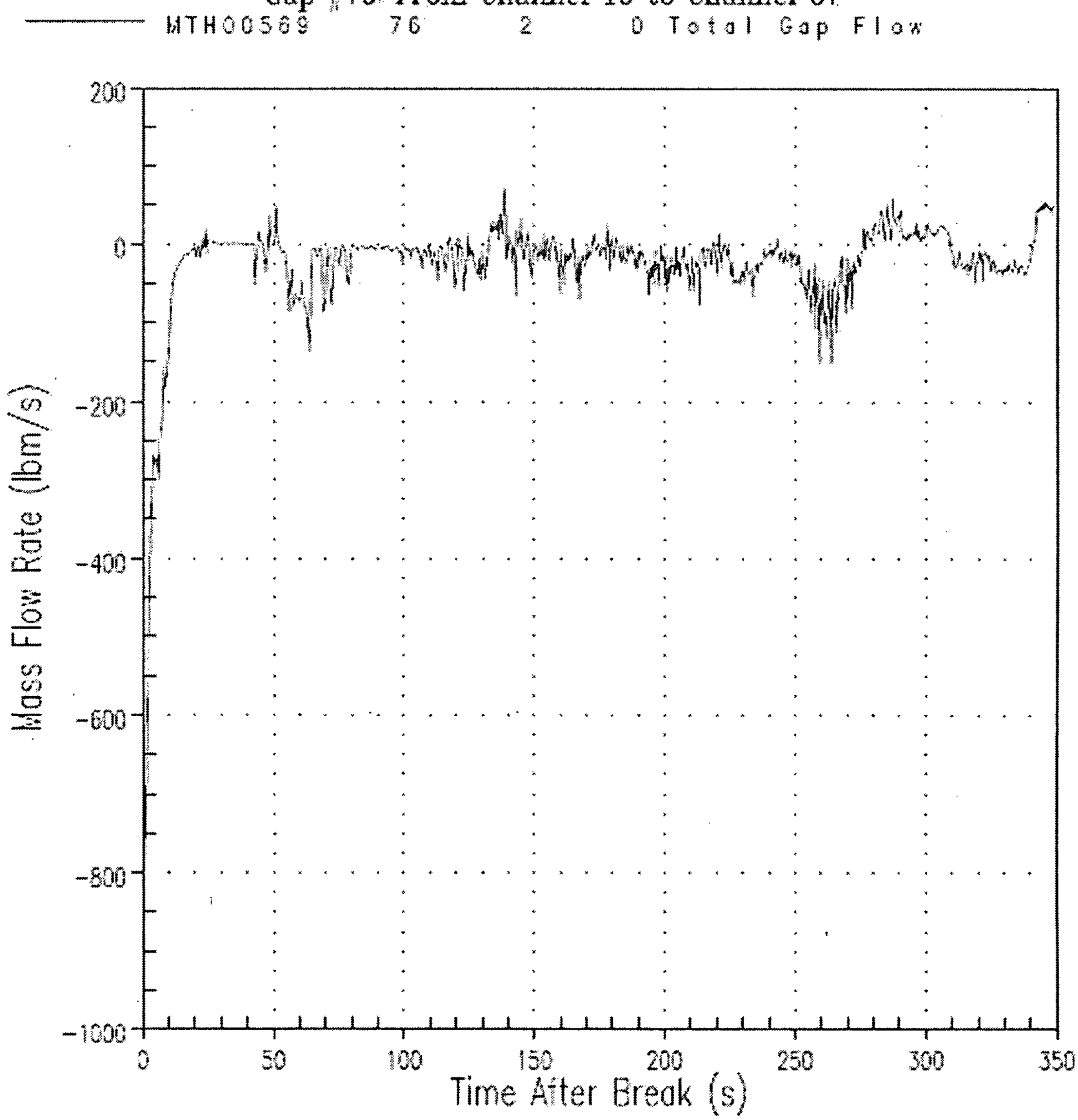
PTN Limiting PCT Case

Figure d51: Total Gap Flow between Intact Loop Downcomer Channels
CCFL – Section 4
Gap #75: From Channel 66 to Channel 67



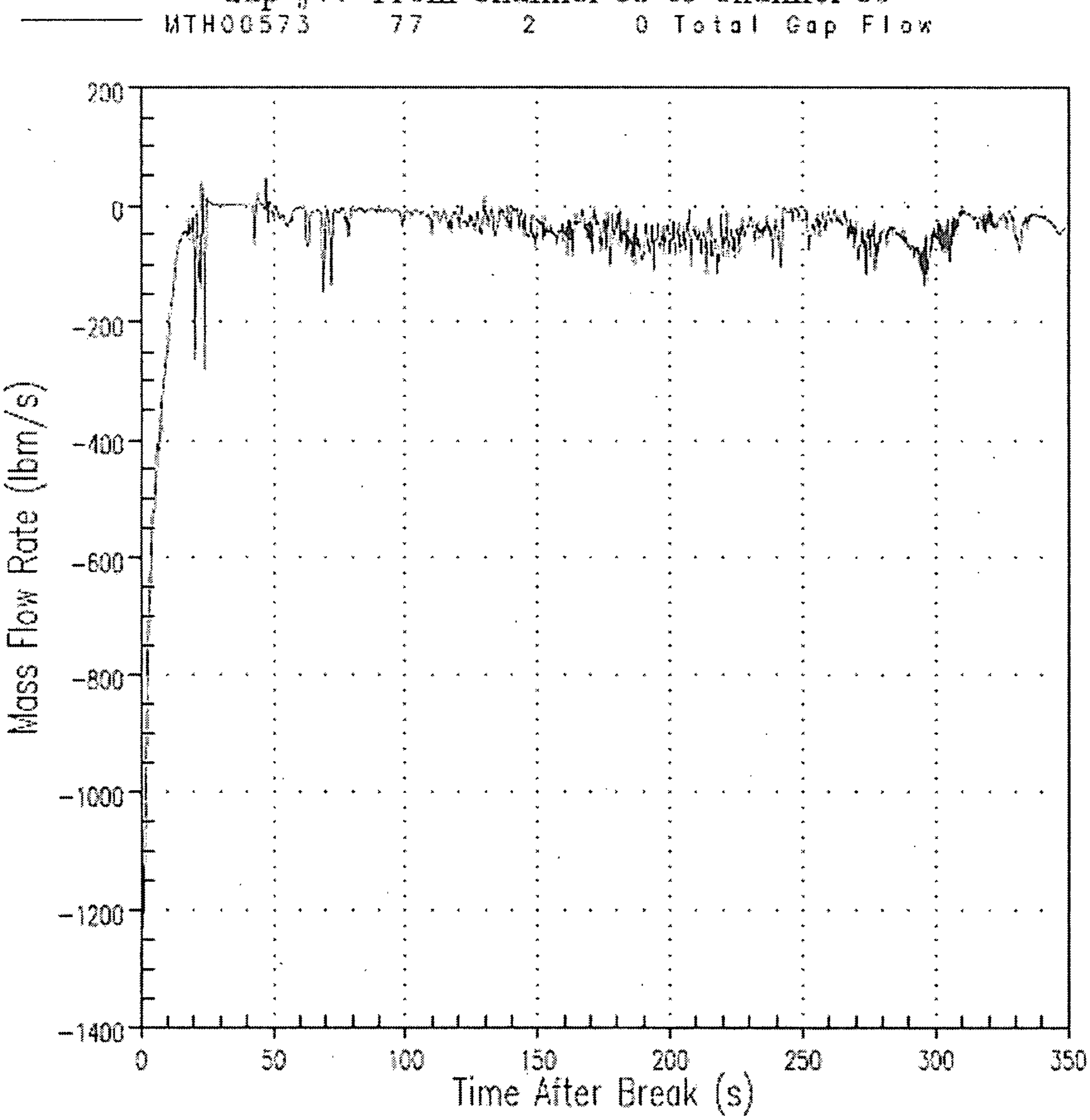
PTN Limiting PCT Case

Figure d52: Total Gap Flow between Intact & Broken Loop Downcomer Channels
CCFL – Section 4
Gap #76: From Channel 16 to Channel 67



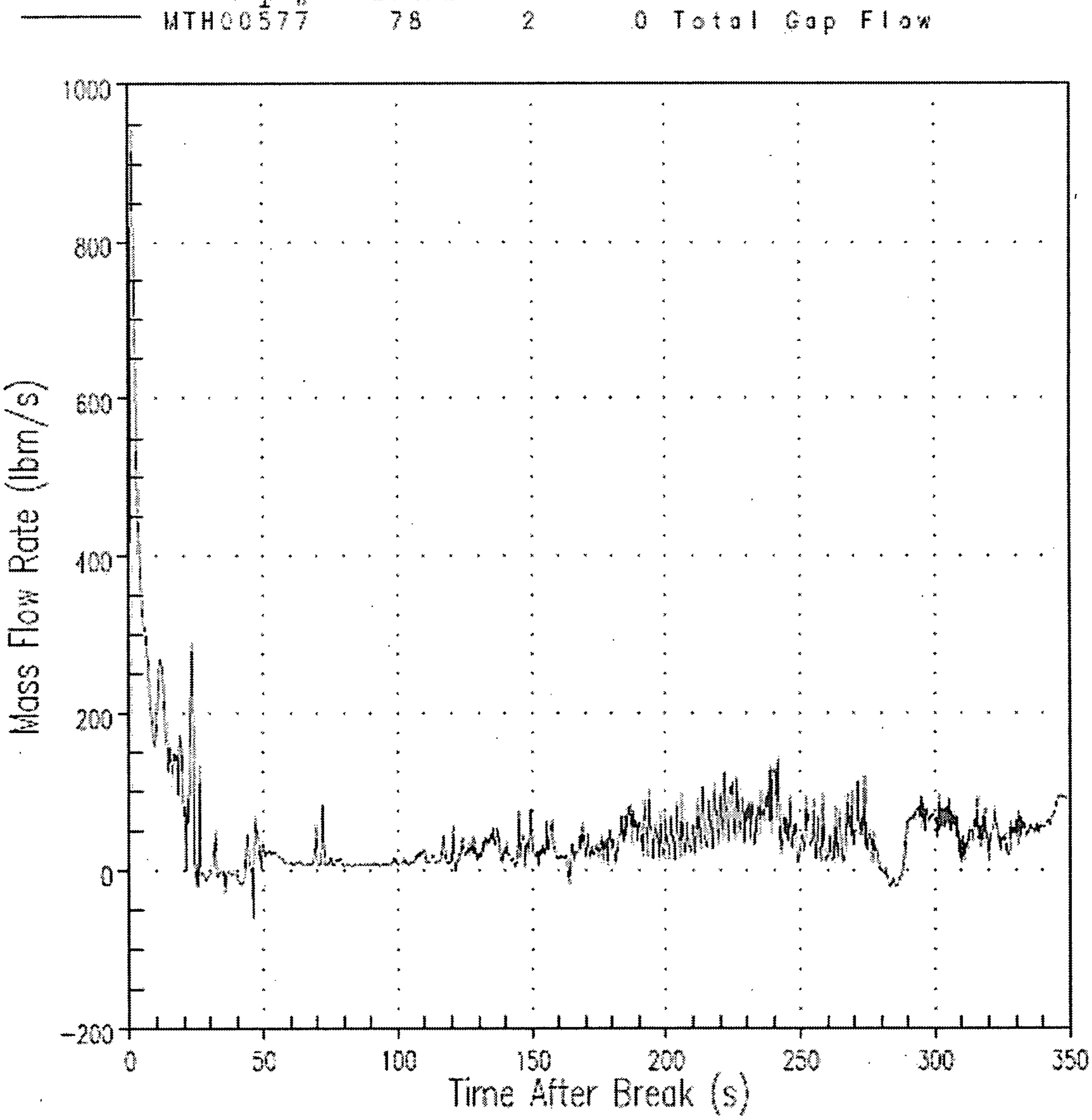
PTN Limiting PCT Case

Figure d53: Total Gap Flow between Intact Loop Downcomer Channels
CCFL – Section 4
Gap #77: From Channel 68 to Channel 69



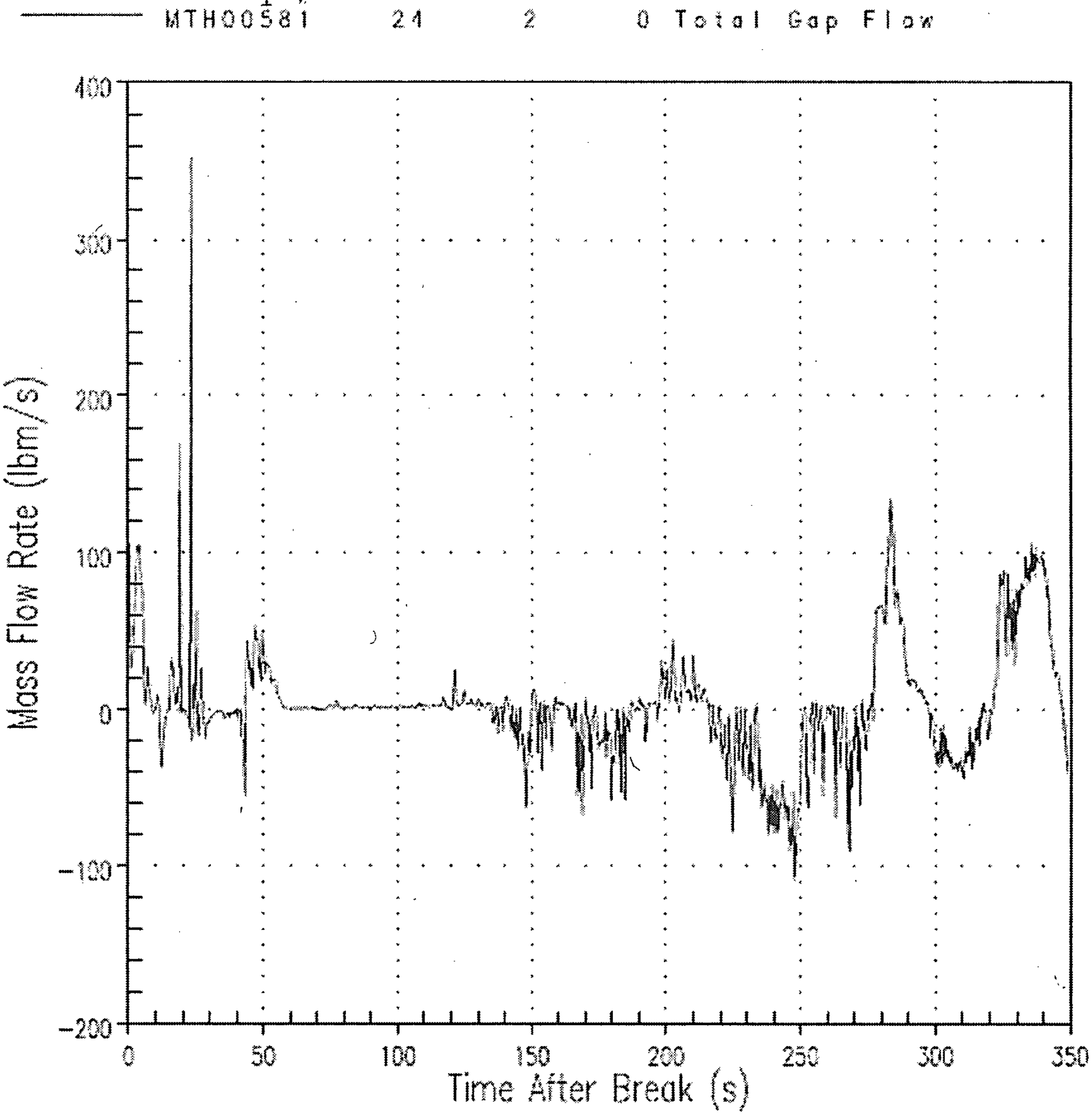
PTN Limiting PCT Case

Figure d54: Total Gap Flow between Intact Loop Downcomer Channels
CCFL - Section 4
Gap #78: From Channel 14 to Channel 69



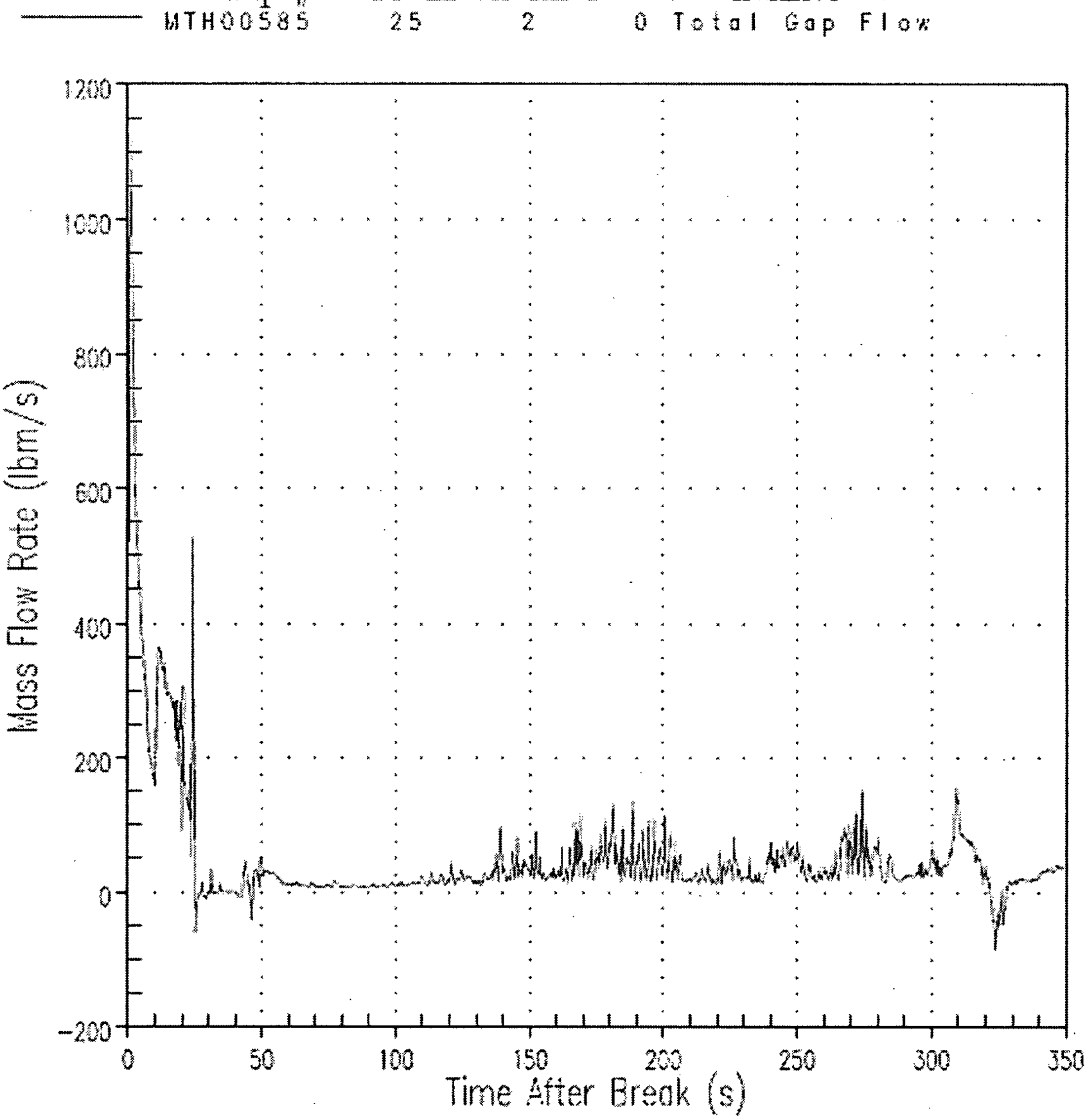
PTN Limiting PCT Case

Figure d55: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum - Section 5 - at Level 1
Gap #24: From Channel 22 to Channel 70



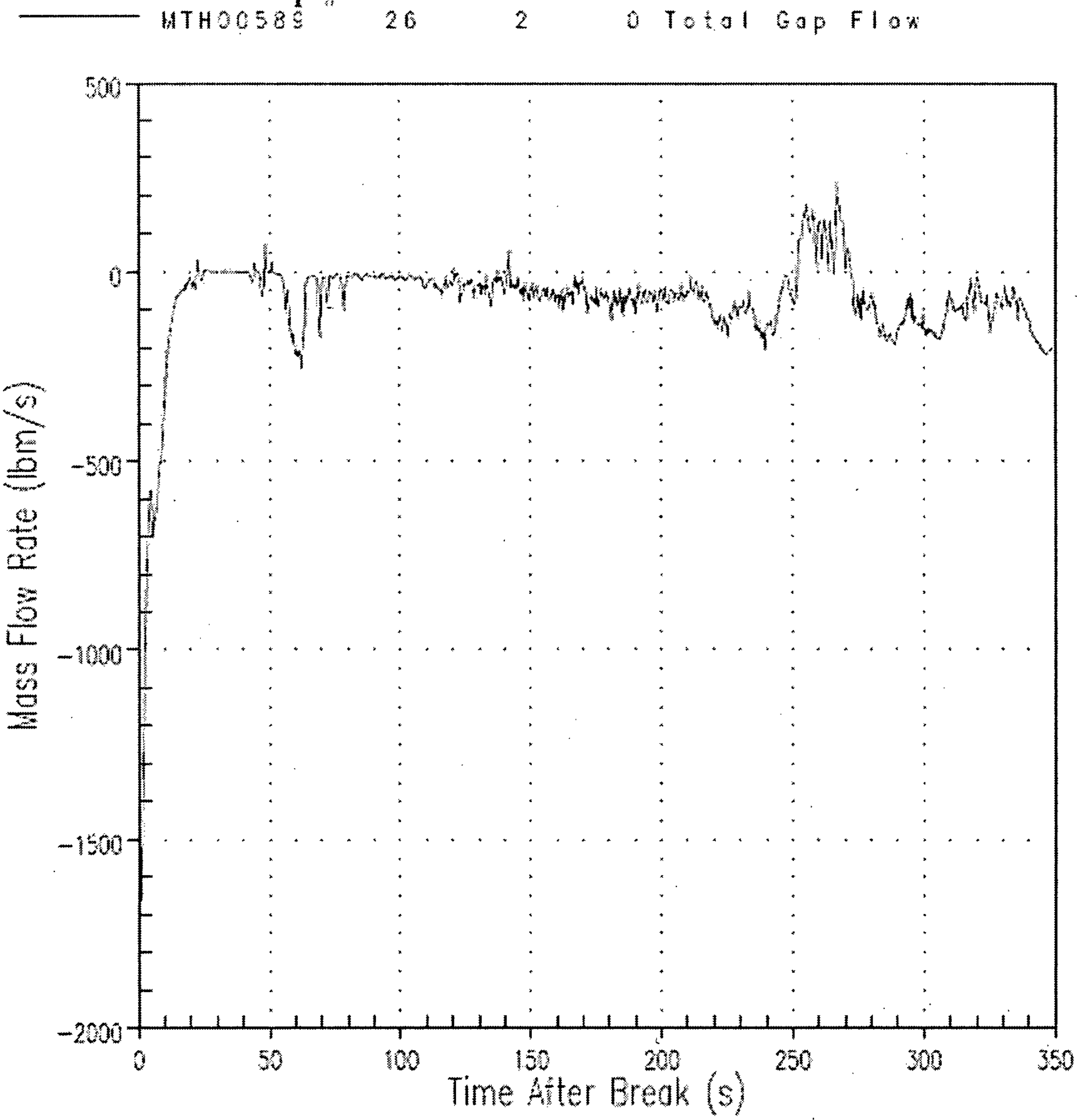
PTN Limiting PCT Case

Figure d56: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum – Section 5 – at Level 1
Gap #25: From Channel 23 to Channel 72



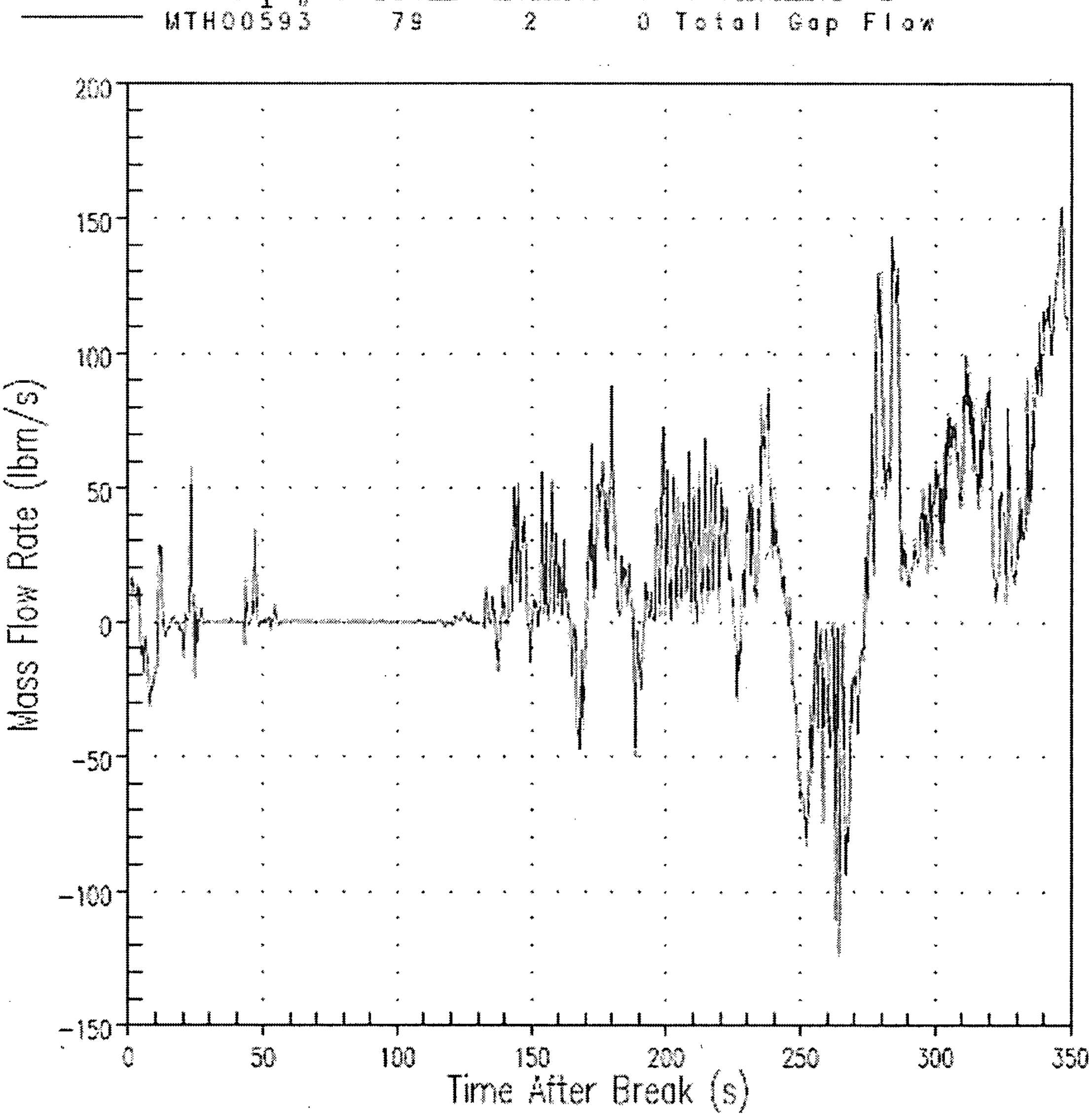
PTN Limiting PCT Case

Figure d57: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Upper Plenum – Section 5 – at Level 1
Gap #26: From Channel 24 to Channel 74



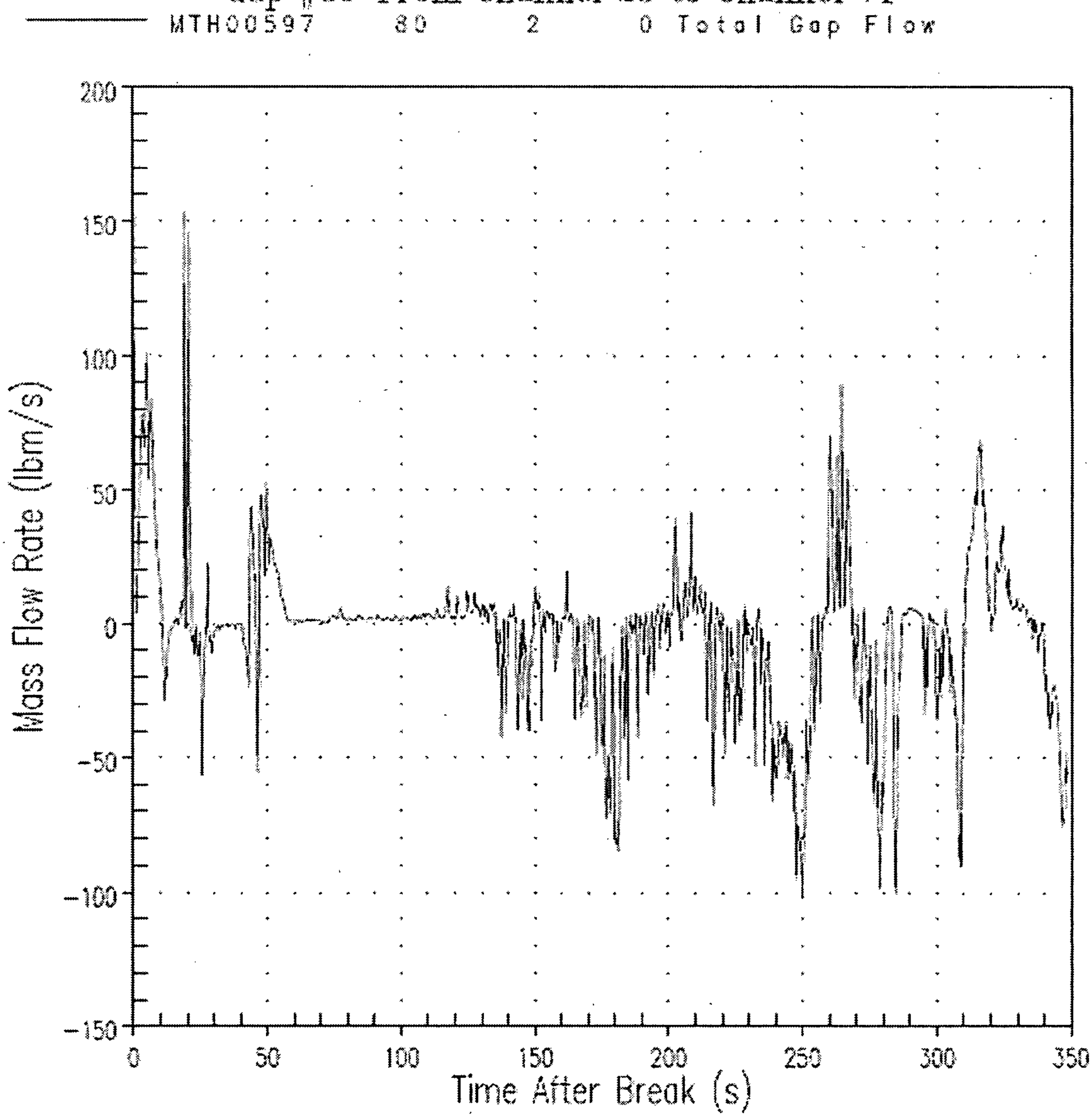
PTN Limiting PCT Case

Figure d58: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum – Section 5 – at Level 1
Gap #79: From Channel 70 to Channel 71



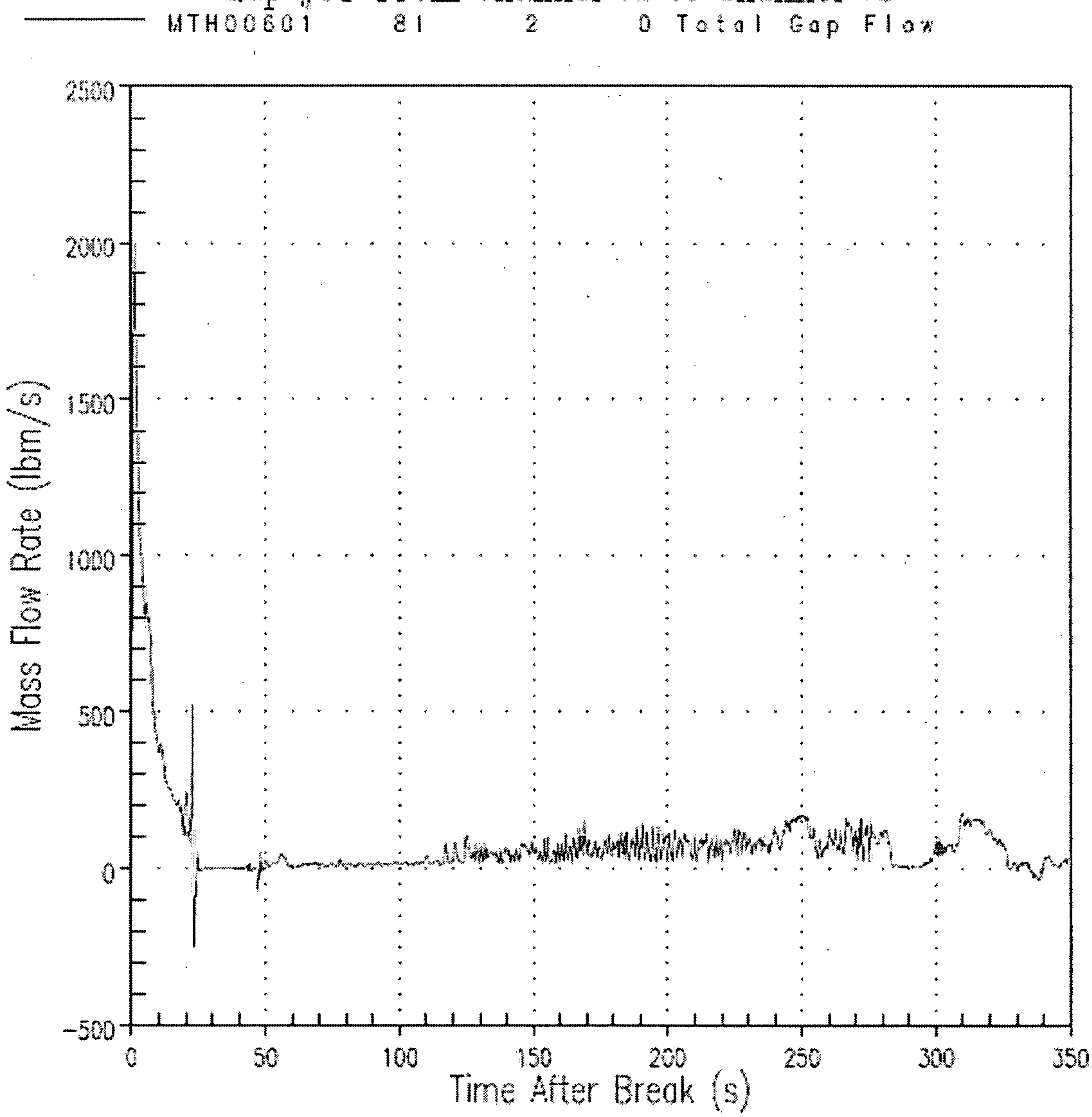
PTN Limiting PCT Case

Figure d59: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum - Section 5 - at Level 1
Gap #80: From Channel 23 to Channel 71



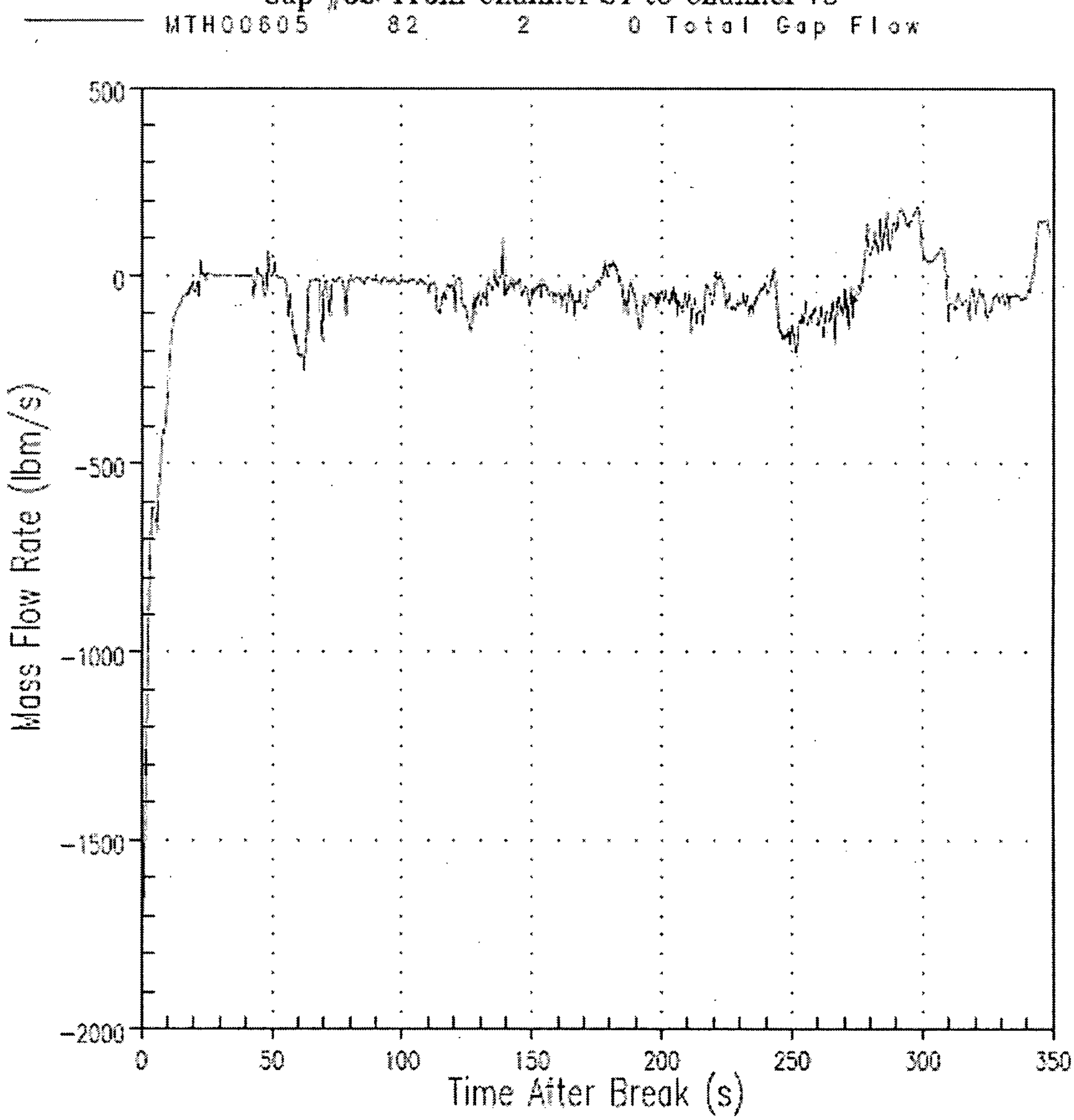
PTN Limiting PCT Case

Figure d60: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum - Section 5 - at Level 1
Gap #81: From Channel 72 to Channel 73



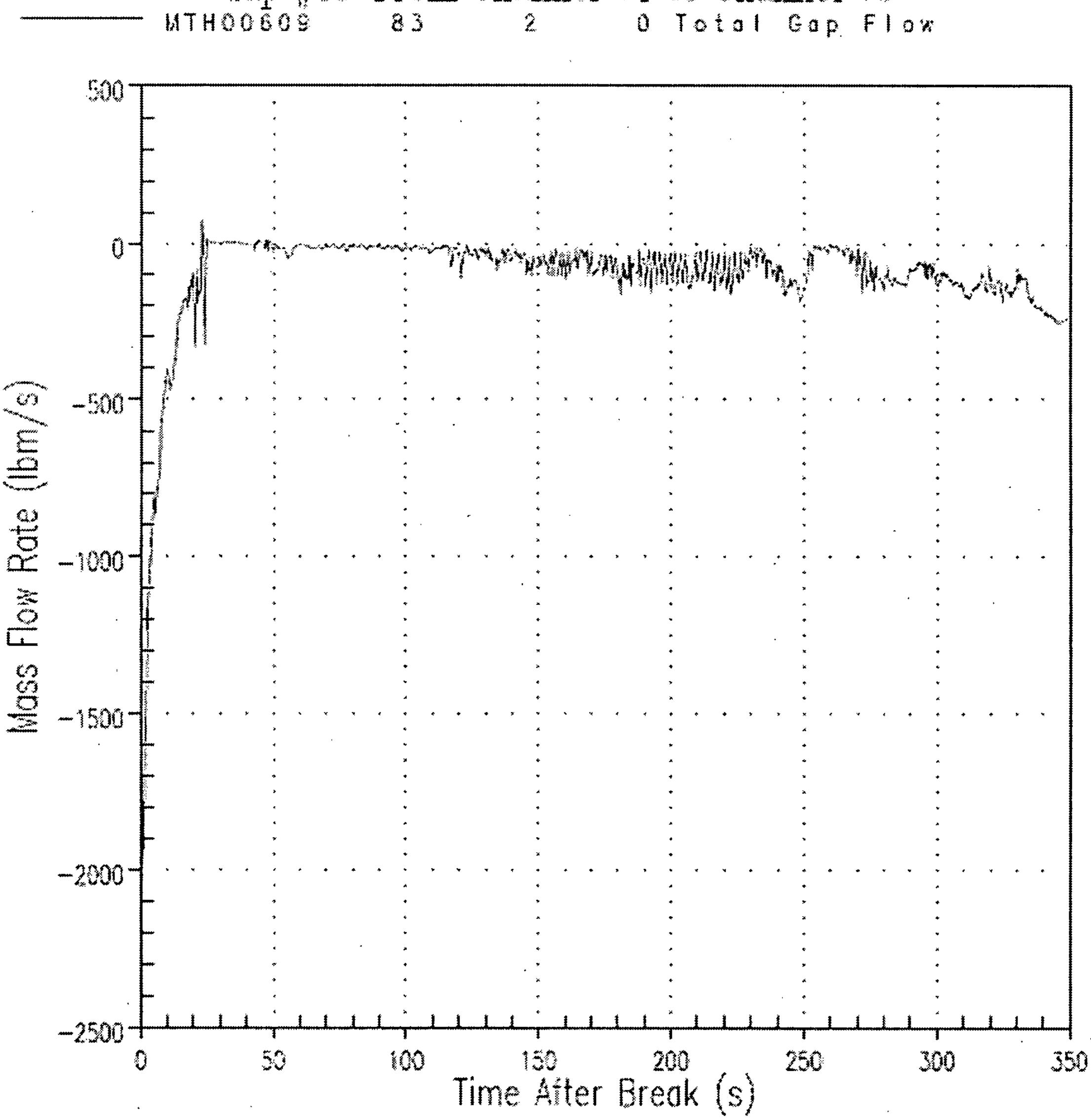
PTN Limiting PCT Case

Figure d61: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Upper Plenum – Section 5 – at Level 1
Gap #82: From Channel 24 to Channel 73



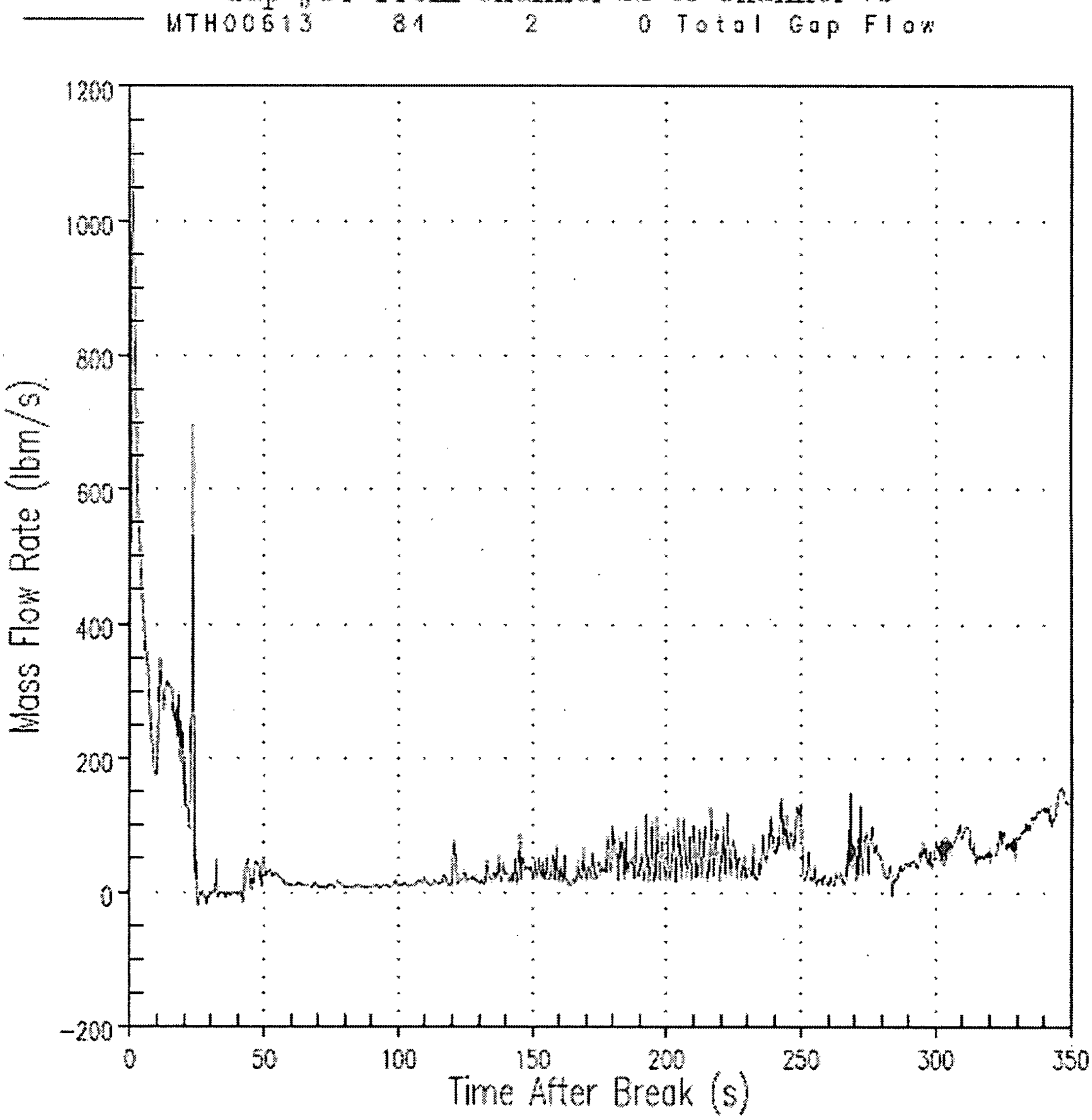
PTN Limiting PCT Case

Figure d62: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum – Section 5 – at Level 1
Gap #83: From Channel 74 to Channel 75



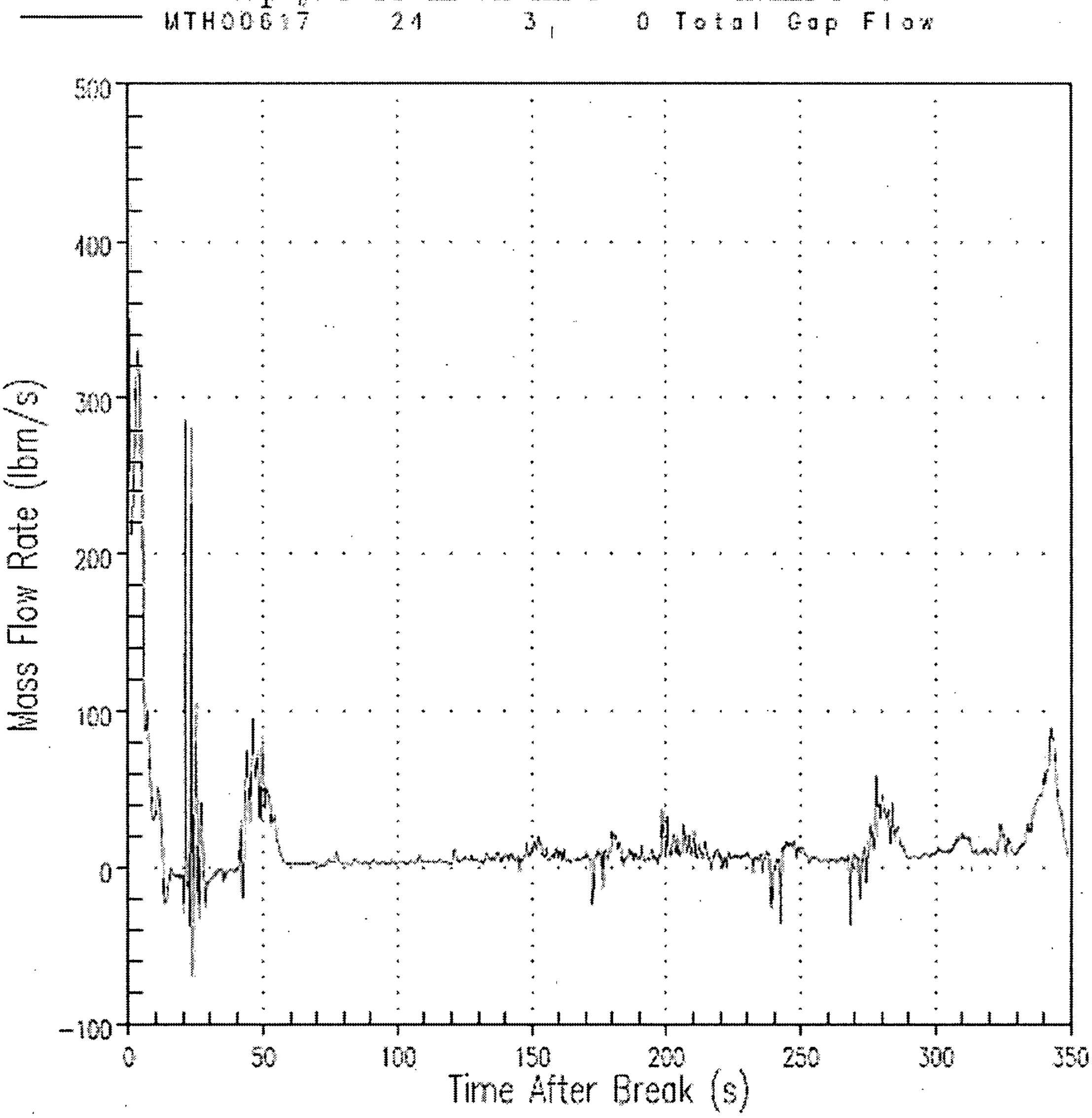
PTN Limiting PCT Case

Figure d63: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum - Section 5 - at Level 1
Gap #84: From Channel 22 to Channel 75



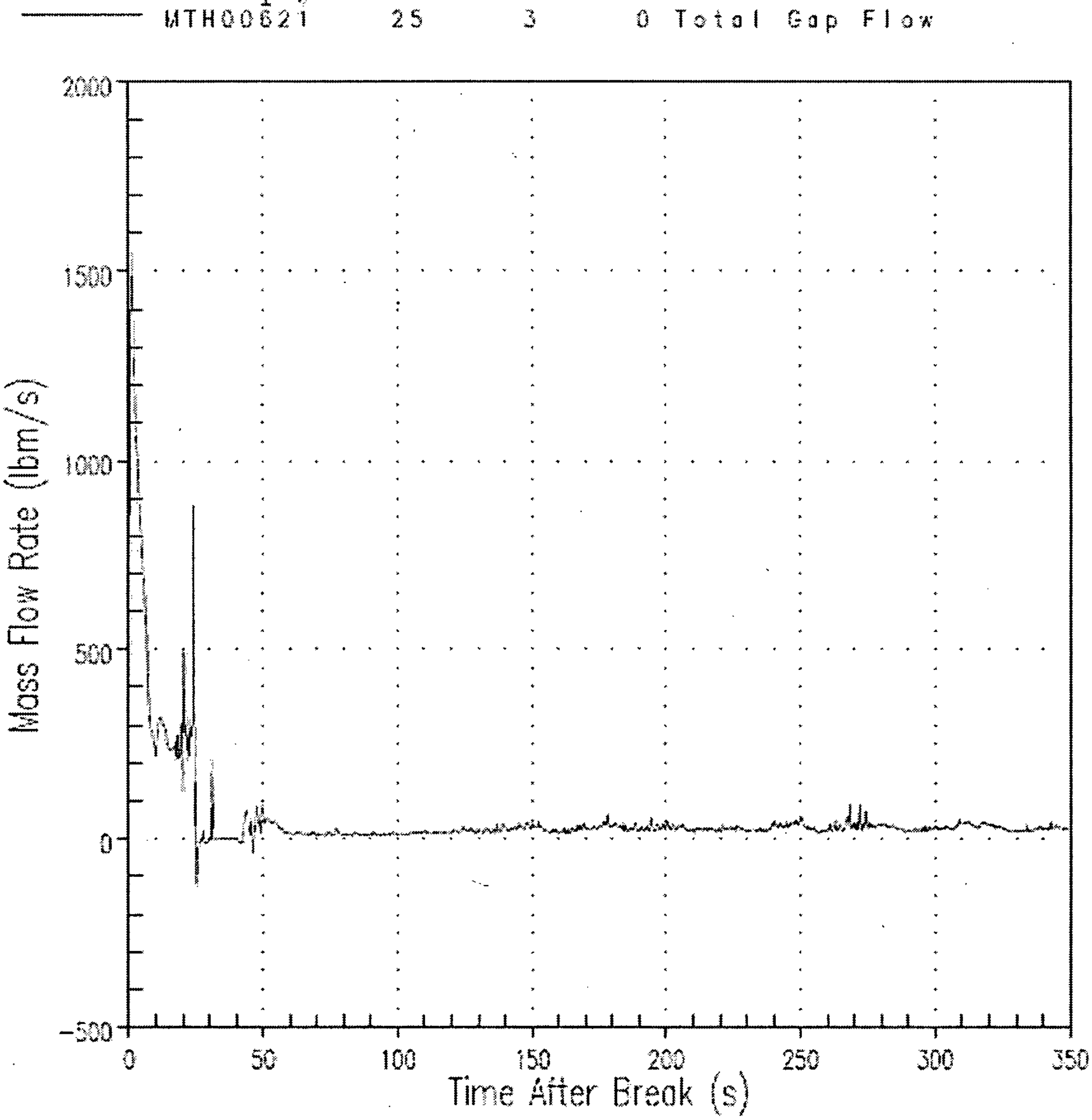
PTN Limiting PCT Case

Figure d64: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum - Section 5 - at Level 2
Gap #24: From Channel 22 to Channel 70



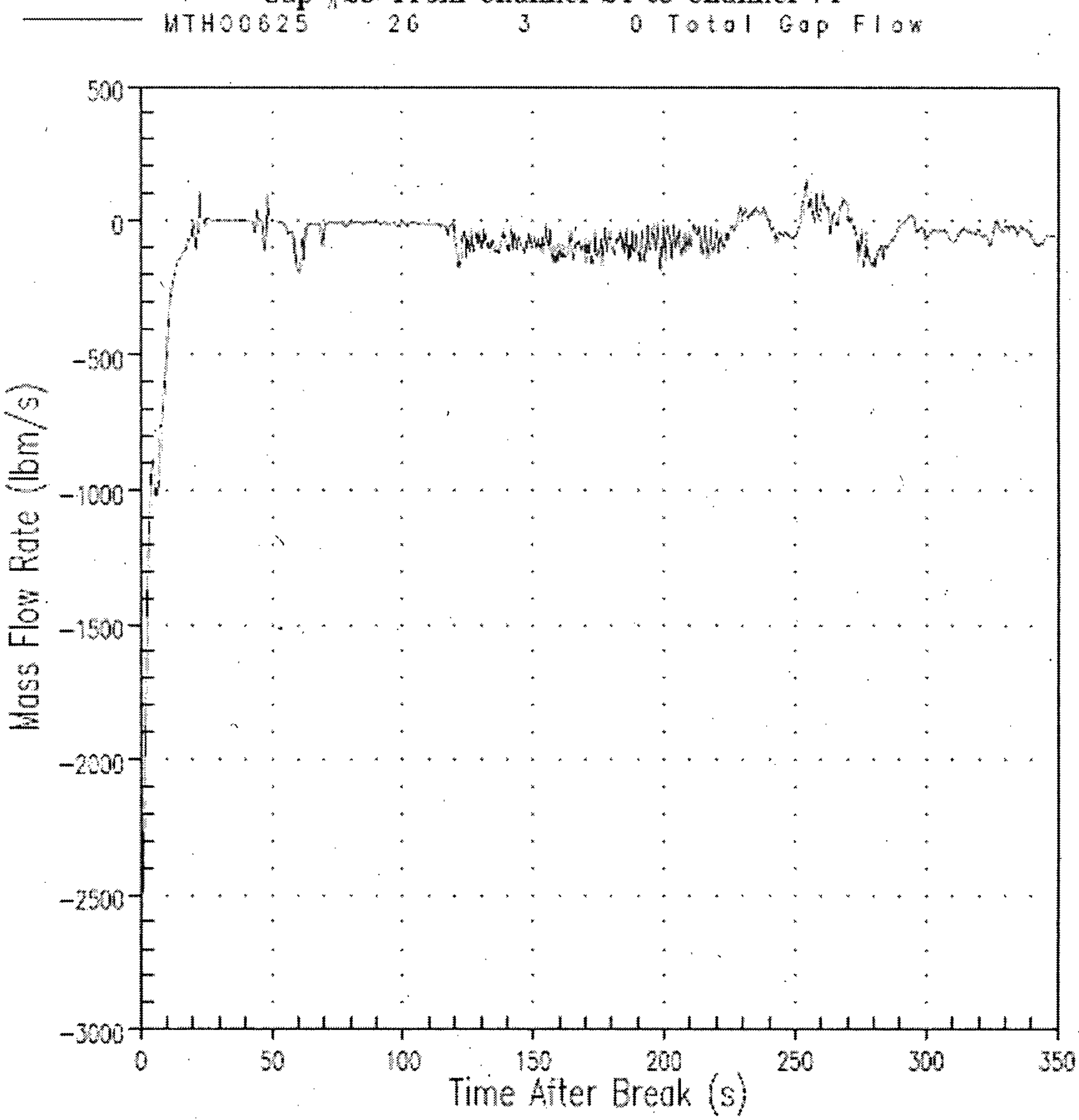
PTN Limiting PCT Case

Figure d65: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum - Section 5 - at Level 2
Gap #25: From Channel 23 to Channel 72



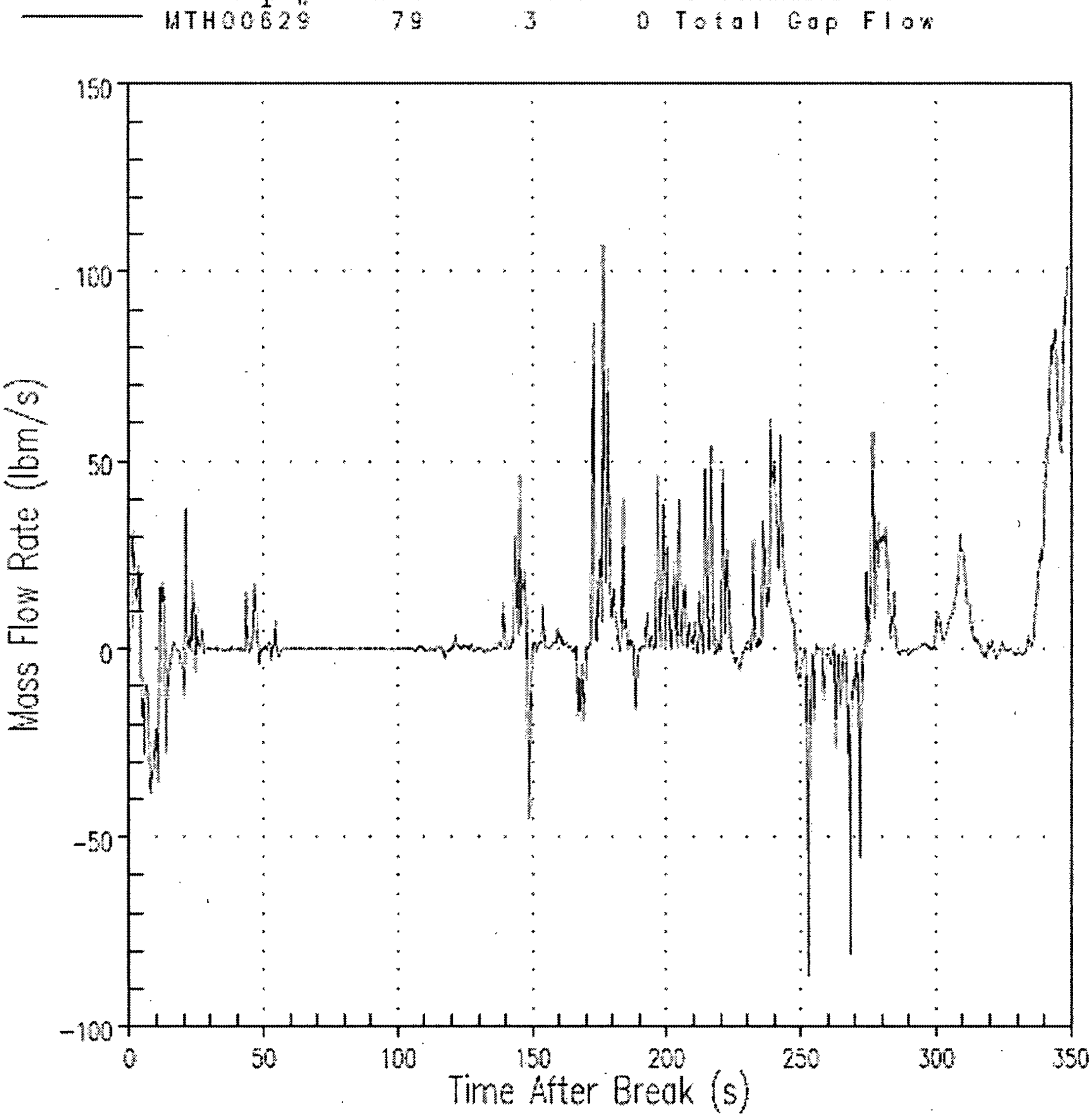
PTN Limiting PCT Case

Figure d66: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Upper Plenum – Section 5 – at Level 2
Gap #26: From Channel 24 to Channel 74



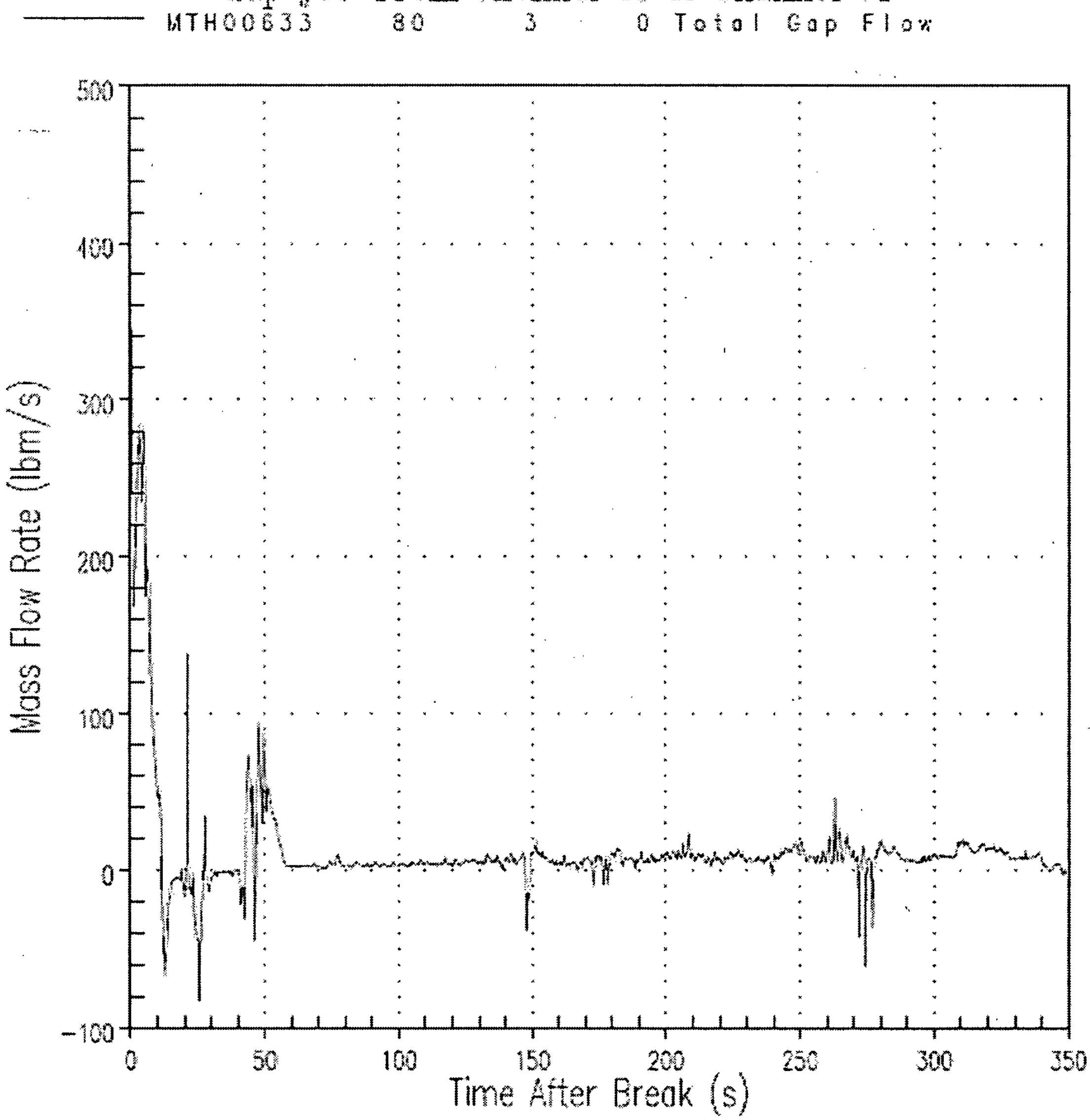
PTN Limiting PCT Case

Figure d67: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum – Section 5 – at Level 2
Gap #79: From Channel 70 to Channel 71



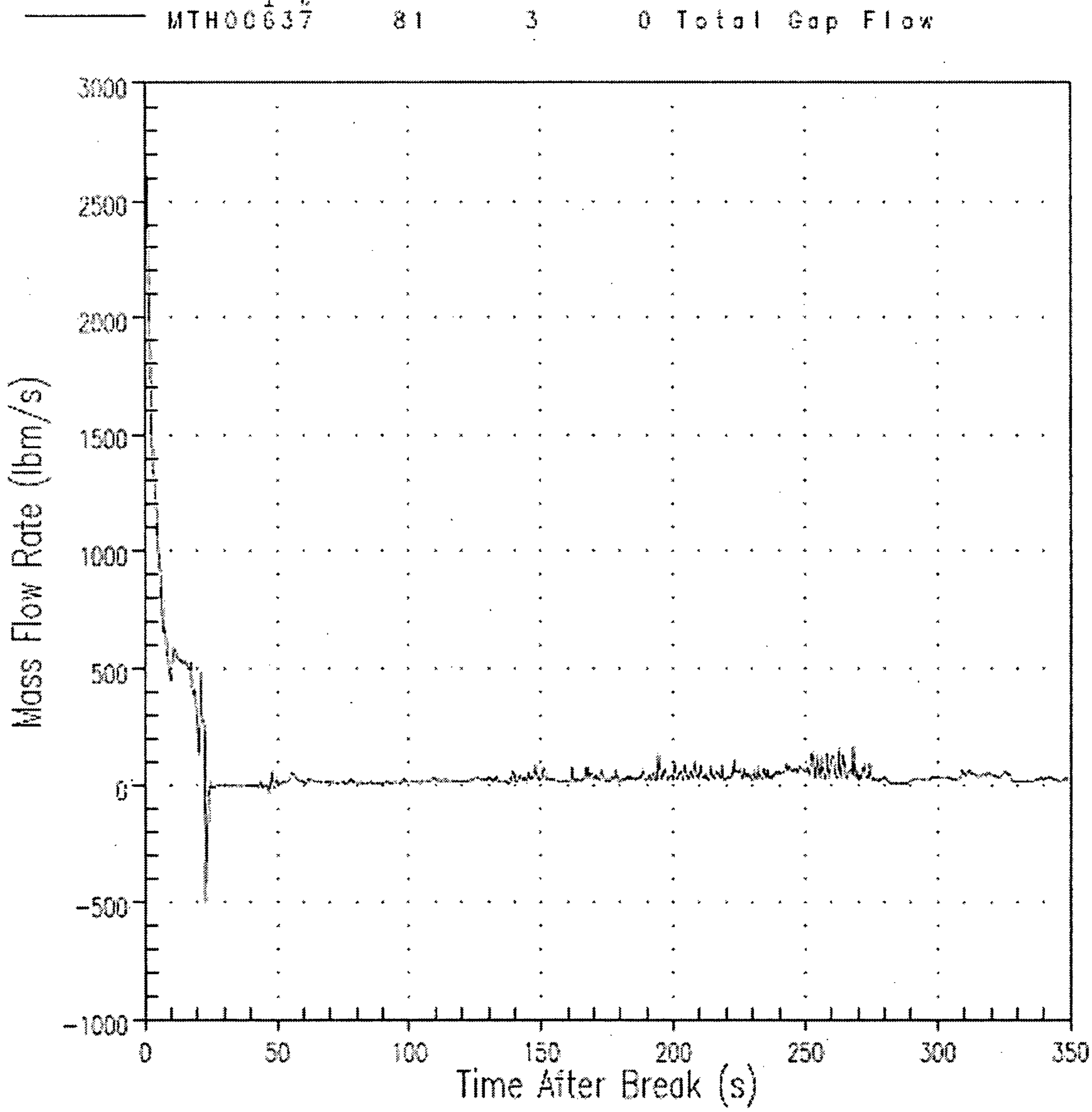
PTN Limiting PCT Case

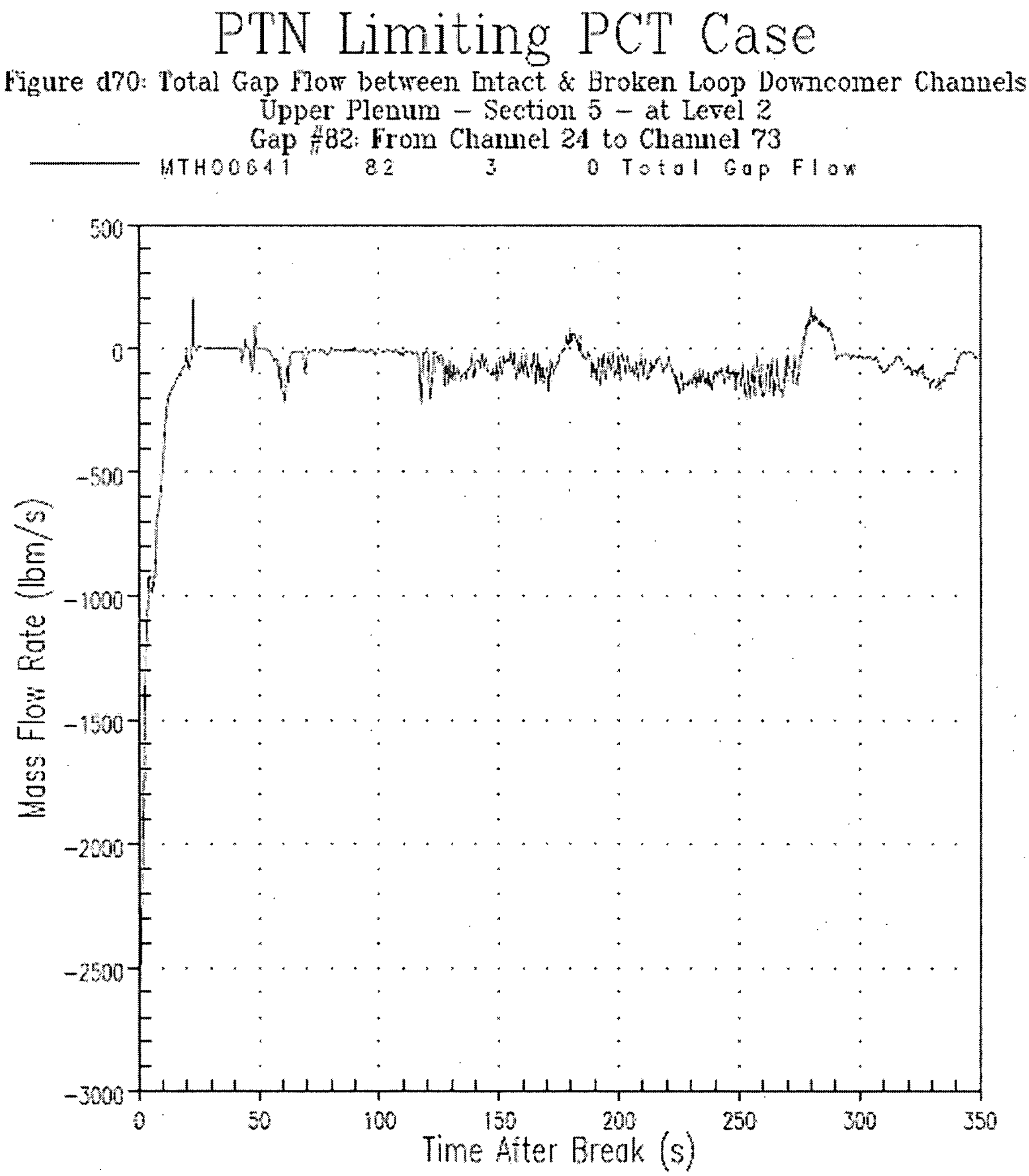
Figure d68: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum – Section 5 – at Level 2
Gap #80: From Channel 23 to Channel 71



PTN Limiting PCT Case

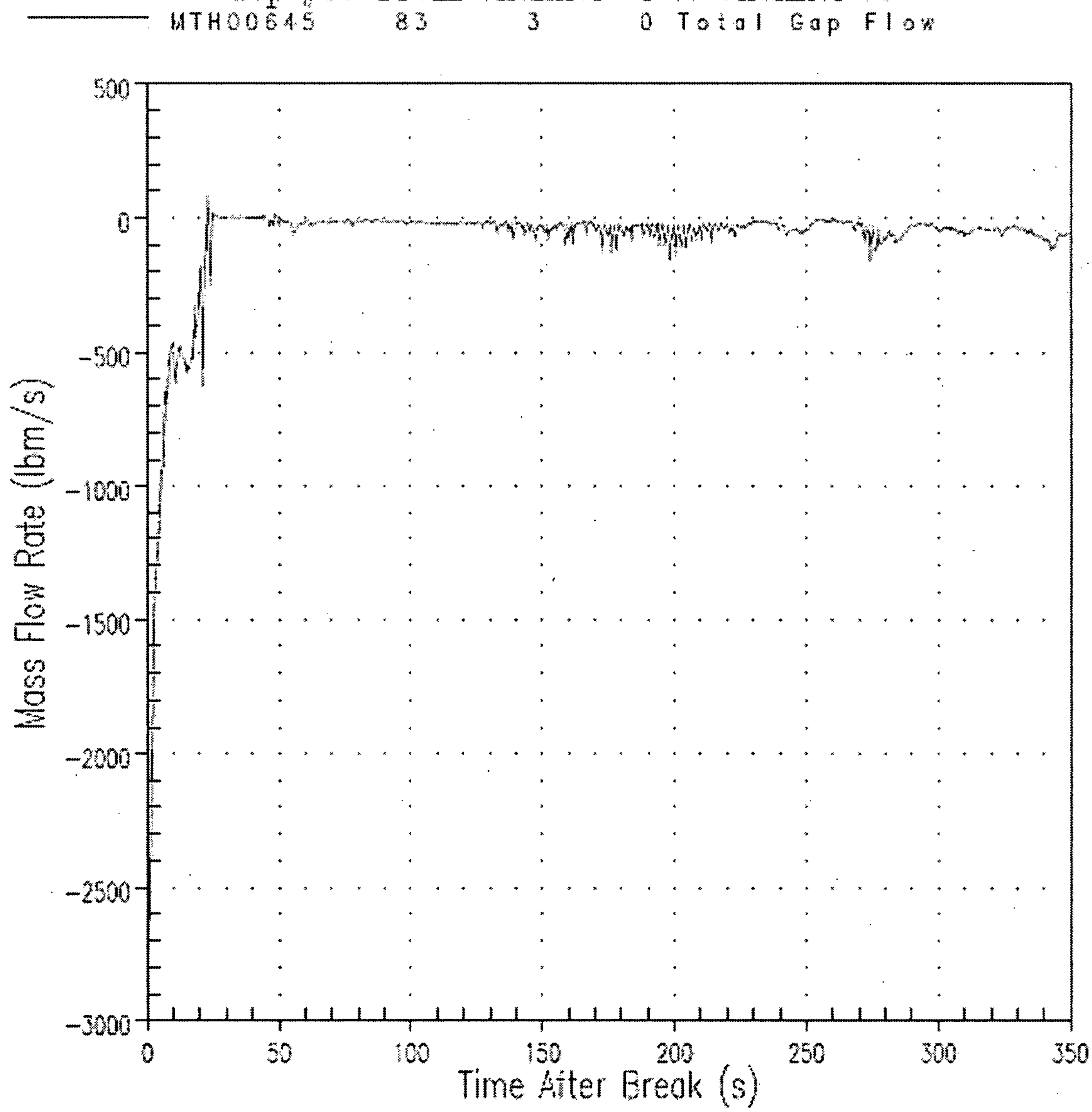
Figure d69: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum - Section 5 - at Level 2
Gap #81: From Channel 72 to Channel 73




PTN Limiting PCT Case
Figure d70: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Upper Plenum – Section 5 – at Level 2
Gap #82: From Channel 24 to Channel 73
MTH00641 82 3 0 Total Gap Flow
Mass Flow Rate (lbm/s)
500
0
-500
-1000
-1500
-2000
-2500
-3000
0
50
100
150
200
250
300
350
Time After Break (s)

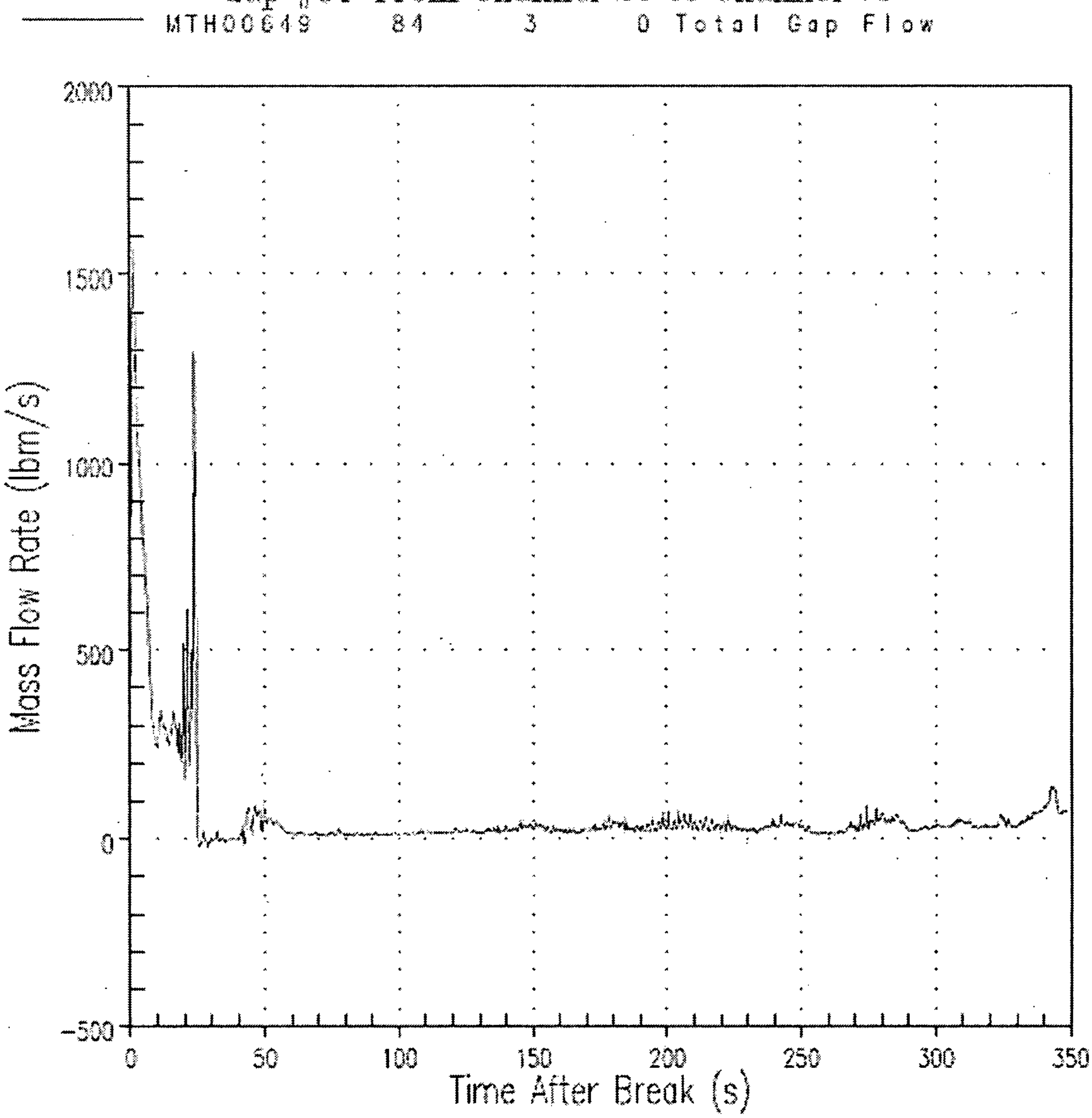
PTN Limiting PCT Case

Figure d71: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum - Section 5 - at Level 2
Gap #83: From Channel 74 to Channel 75



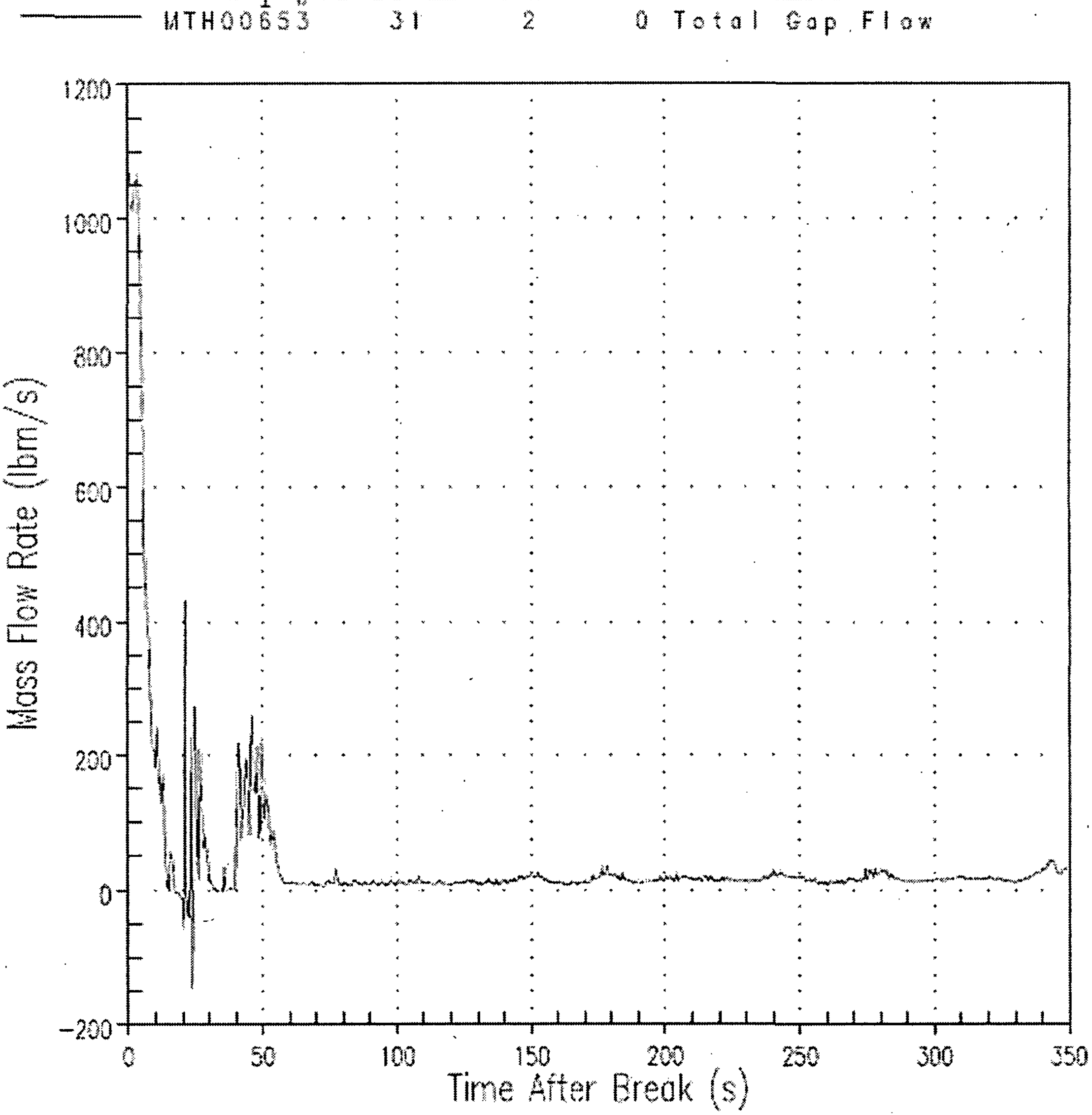
PTN Limiting PCT Case

Figure d72: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum - Section 5 - at Level 2
Gap #84: From Channel 22 to Channel 75



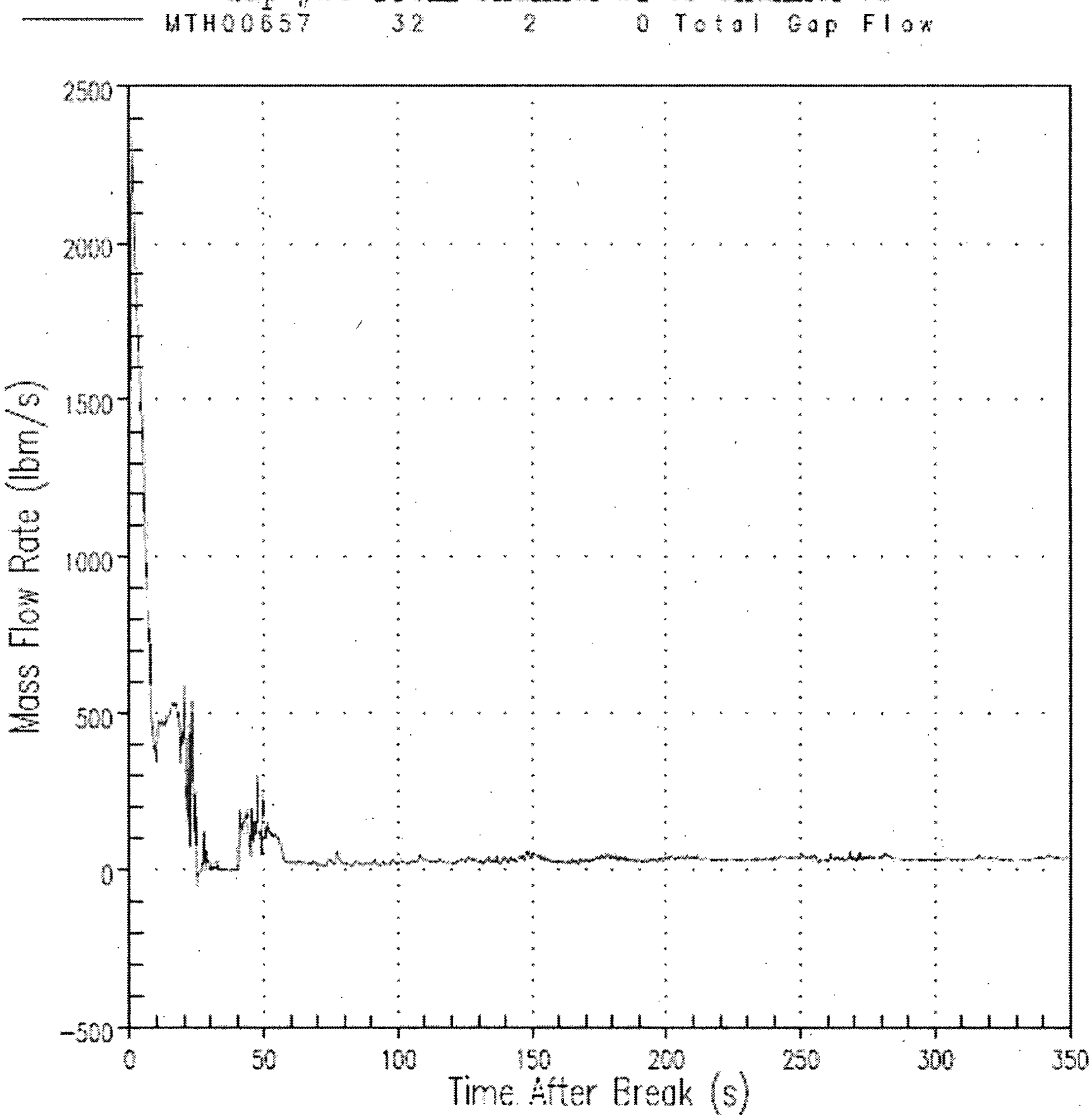
PTN Limiting PCT Case

Figure d73: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum – Section 6
Gap #31: From Channel 30 to Channel 76



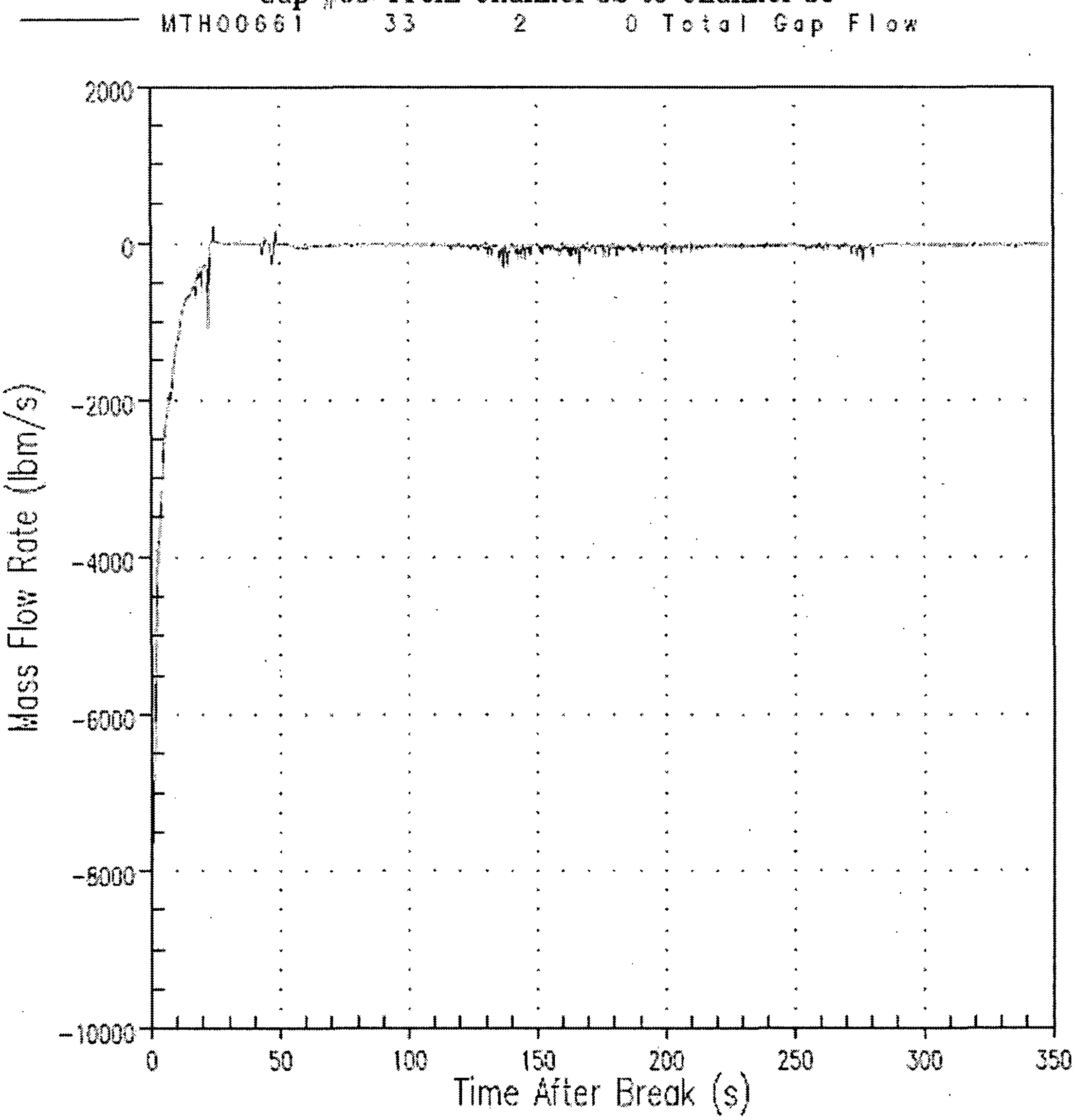
PTN Limiting PCT Case

Figure d74: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum – Section 6
Gap #32: From Channel 31 to Channel 78



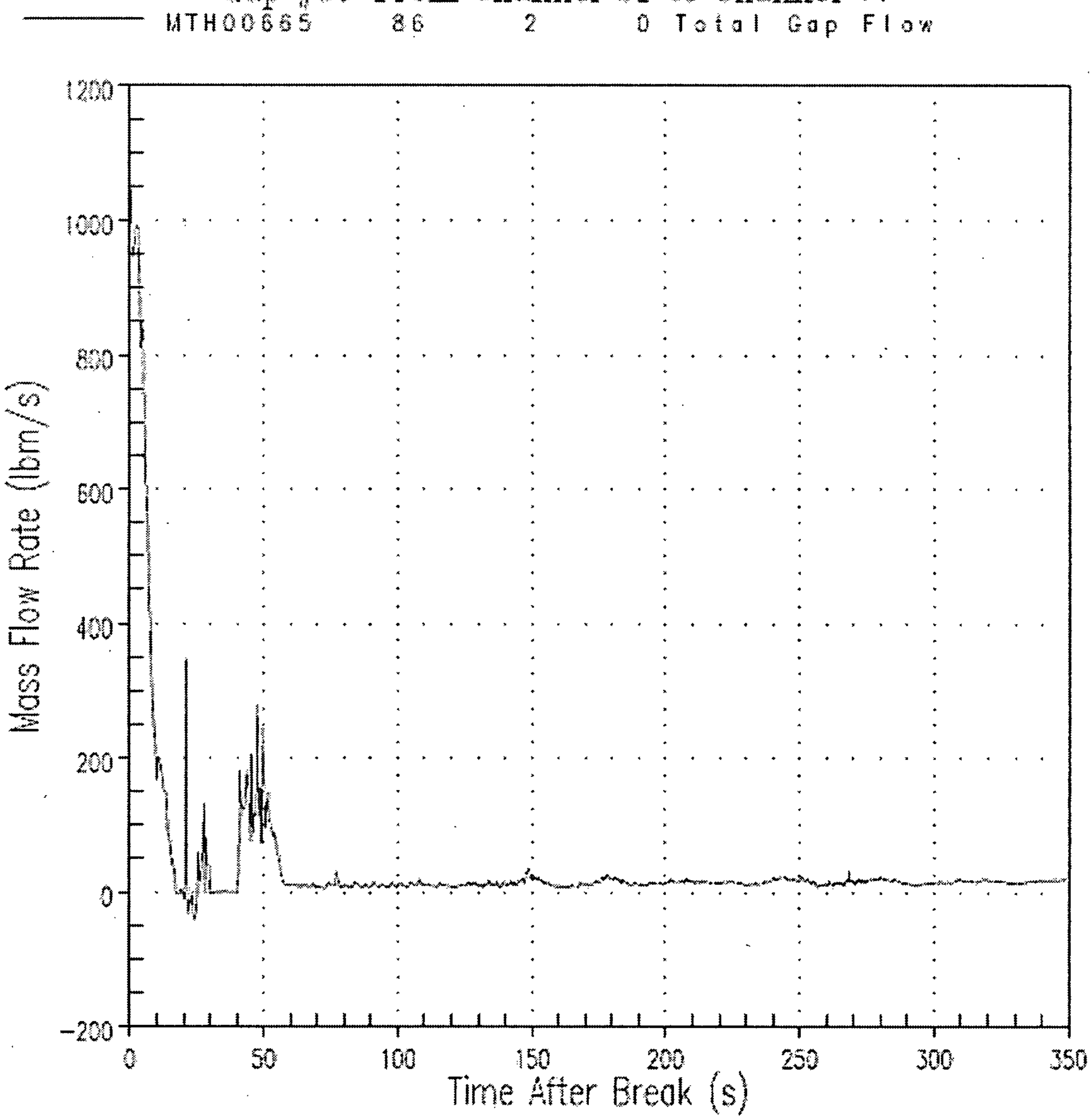
PTN Limiting PCT Case

Figure d75: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Upper Plenum – Section 6
Gap #33: From Channel 32 to Channel 80



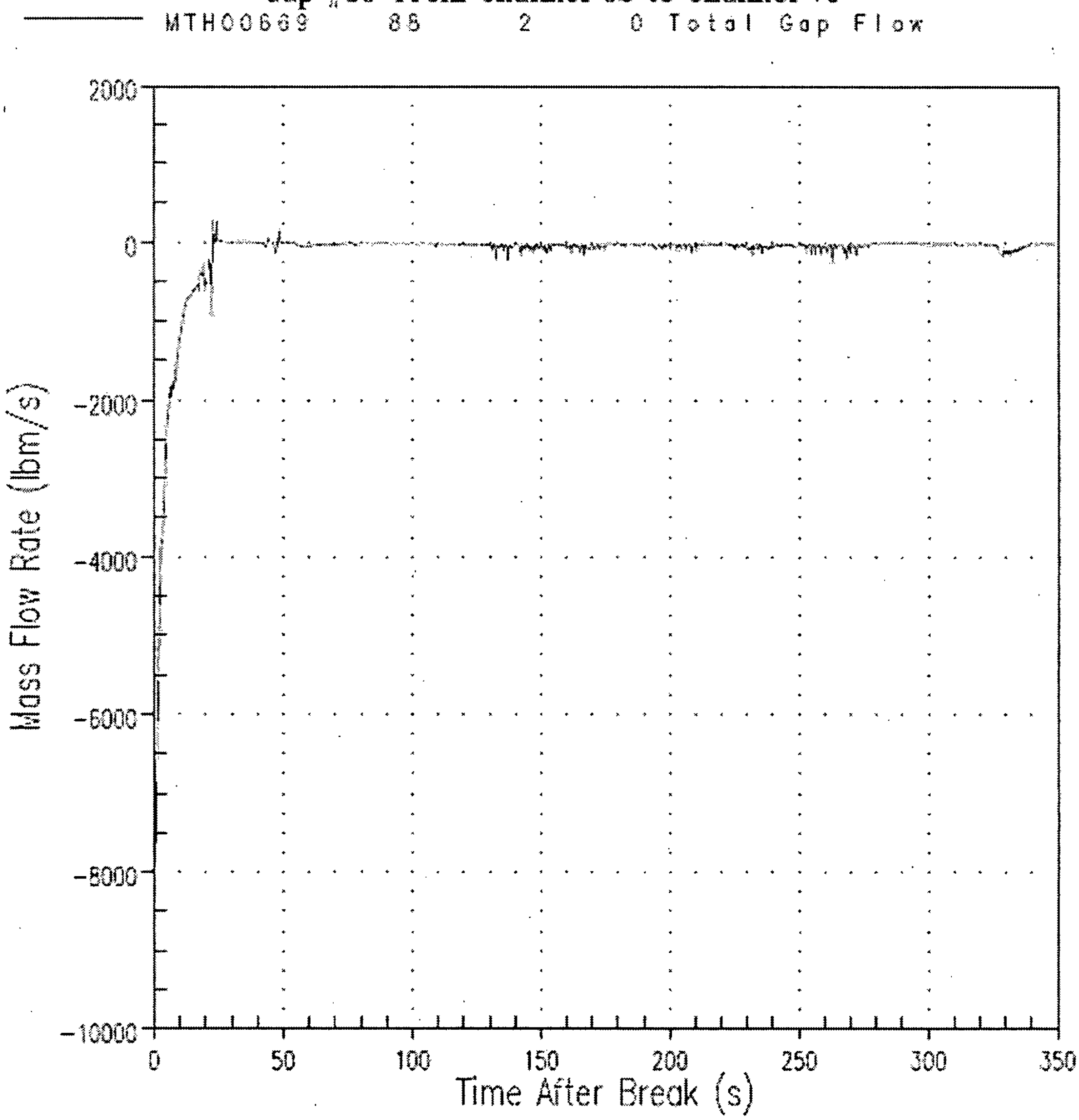
PTN Limiting PCT Case

Figure d76: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum – Section 6
Gap #86: From Channel 31 to Channel 77



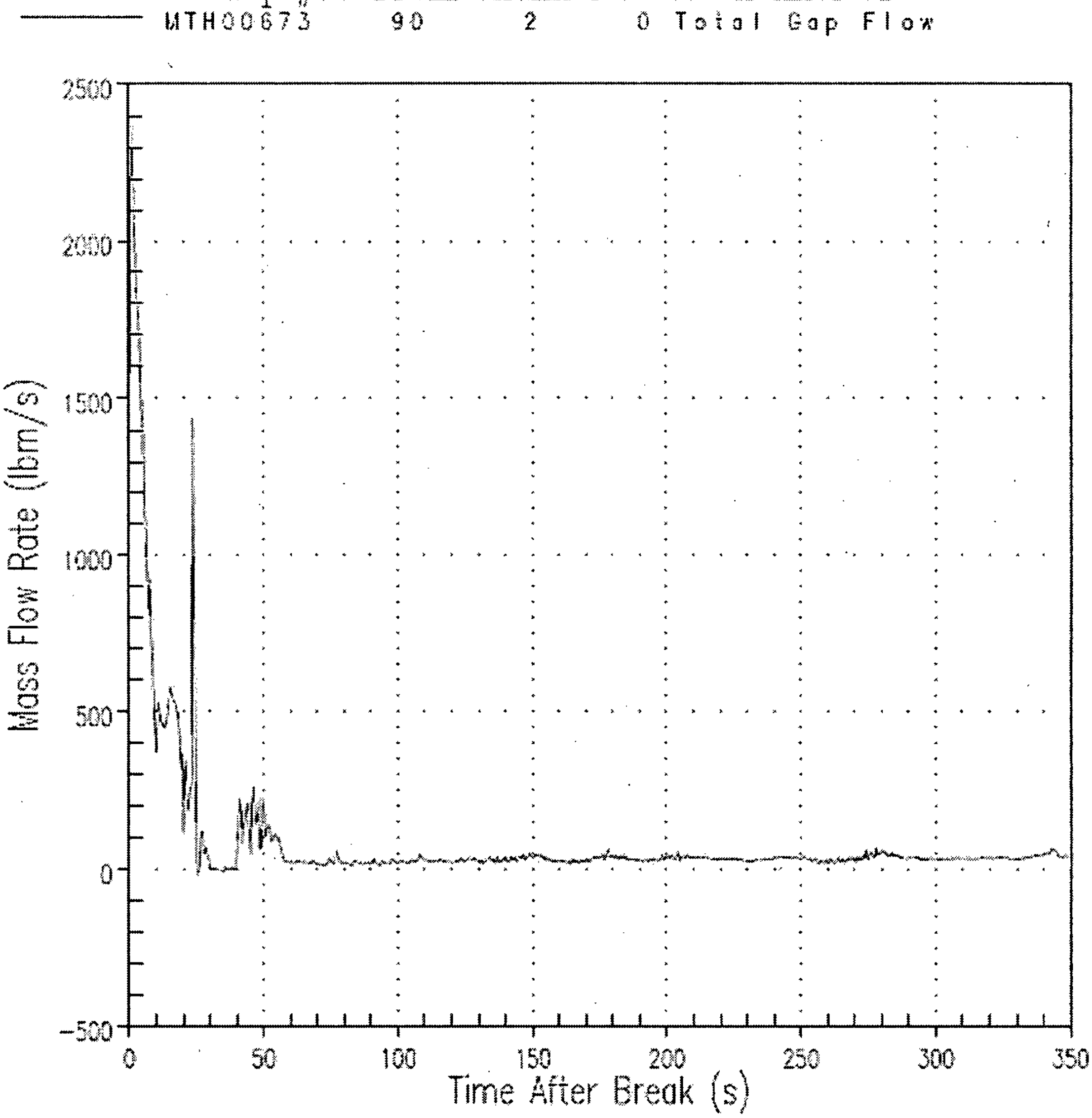
PTN Limiting PCT Case

Figure d77: Total Gap Flow between Intact & Broken Loop Downcomer Channels
Upper Plenum – Section 6
Gap #88: From Channel 32 to Channel 79



PTN Limiting PCT Case

Figure d78: Total Gap Flow between Intact Loop Downcomer Channels
Upper Plenum – Section 6
Gap #90: From Channel 30 to Channel 81



2.8.5.6.3.3 Small Break LOCA

SRXB-1.3.37 Provide plots of core mixture level, pressurizer pressure, and PCT as functions of time that include traces of the 2-, 3-, 4-, and 6-inch break sizes.

The requested plots are provided in Figures 1.3.37-1 – 1.3.37-12 below.

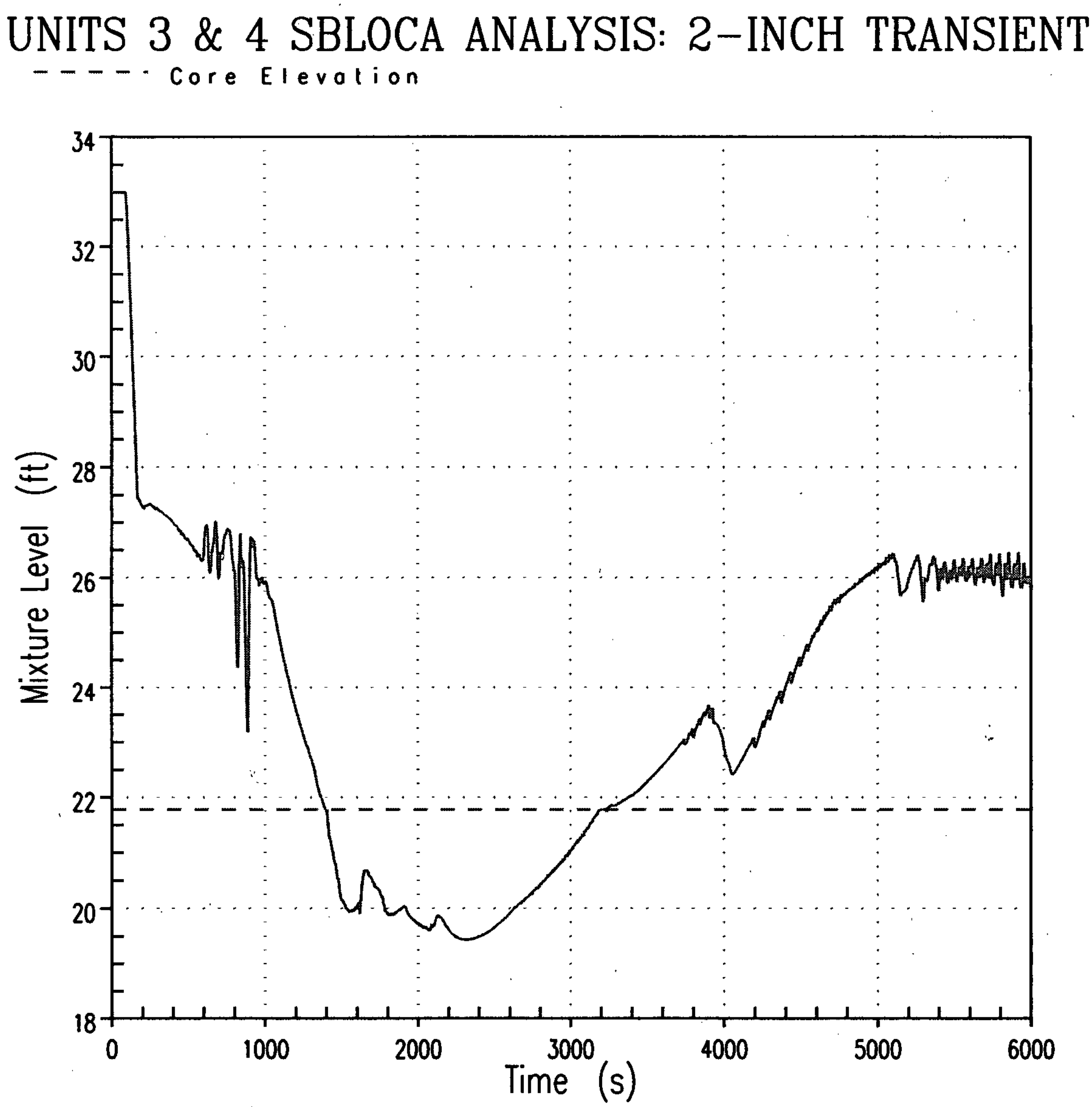


Figure 1.3.37-1: 2-Inch Break – Core Mixture Level

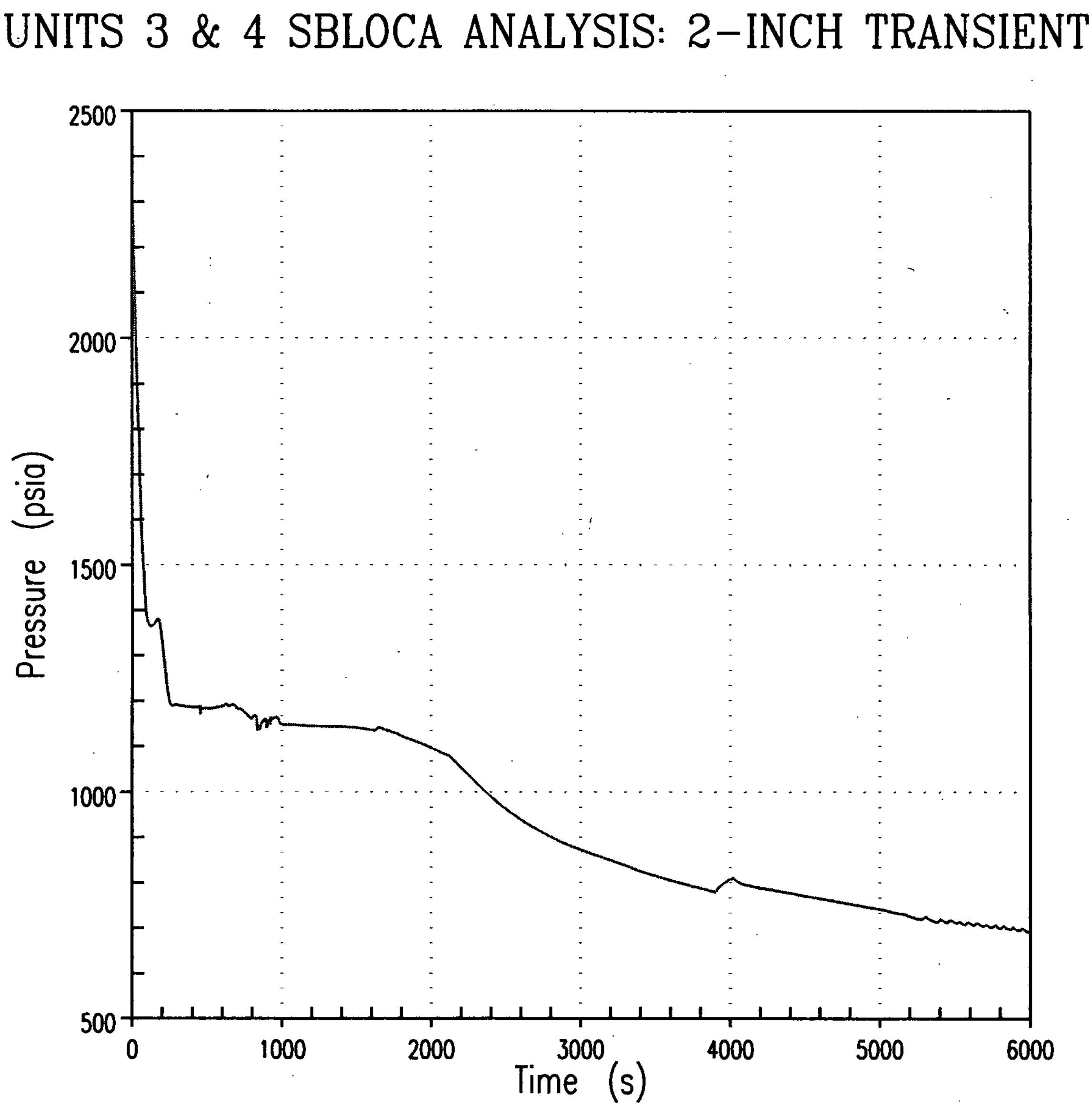


Figure 1.3.37-2: 2-Inch Break – Pressurizer Pressure

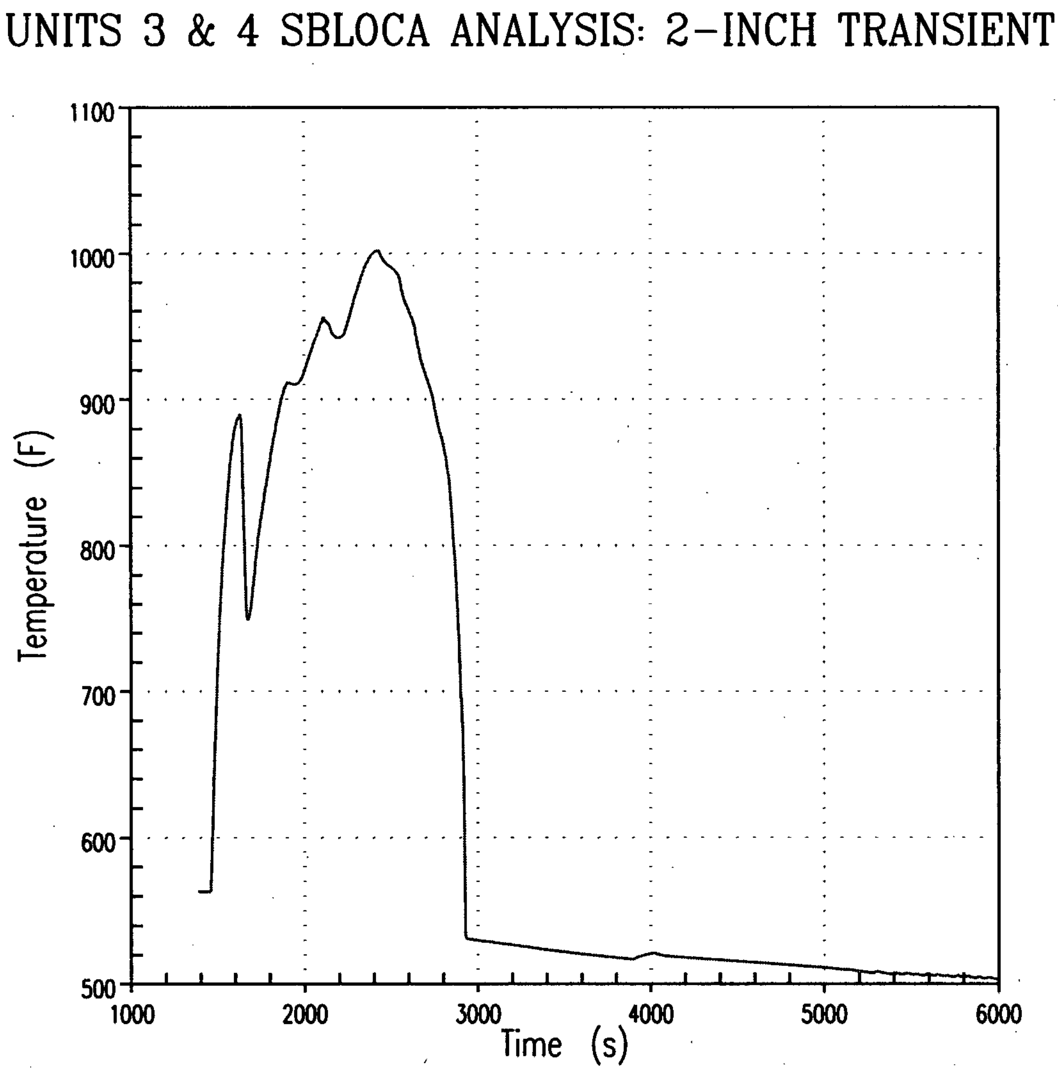


Figure 1.3.37-3: 2-Inch Break – Cladding Temperature at PCT Elevation

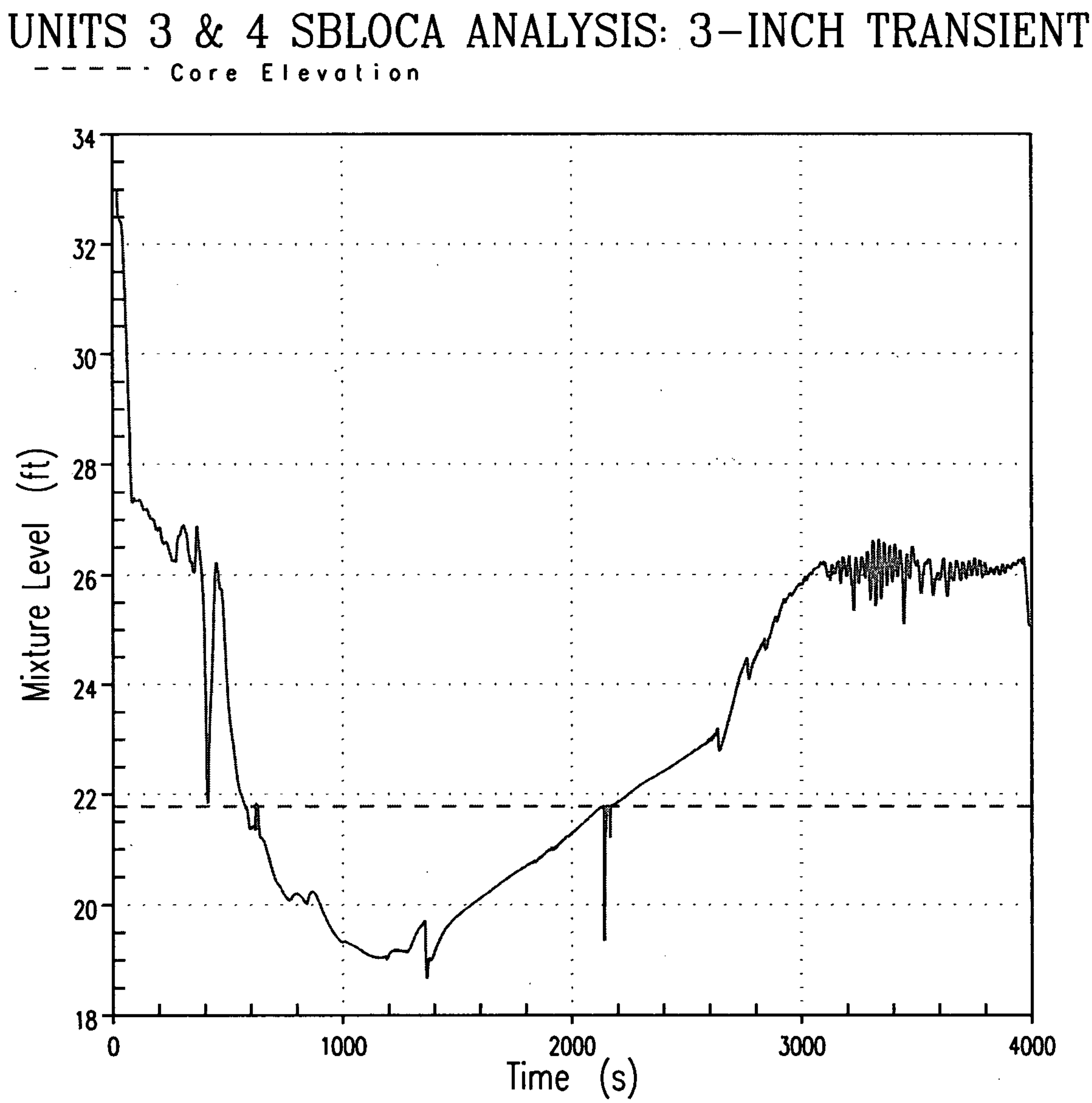


Figure 1.3.37-4: 3-Inch Break – Core Mixture Level

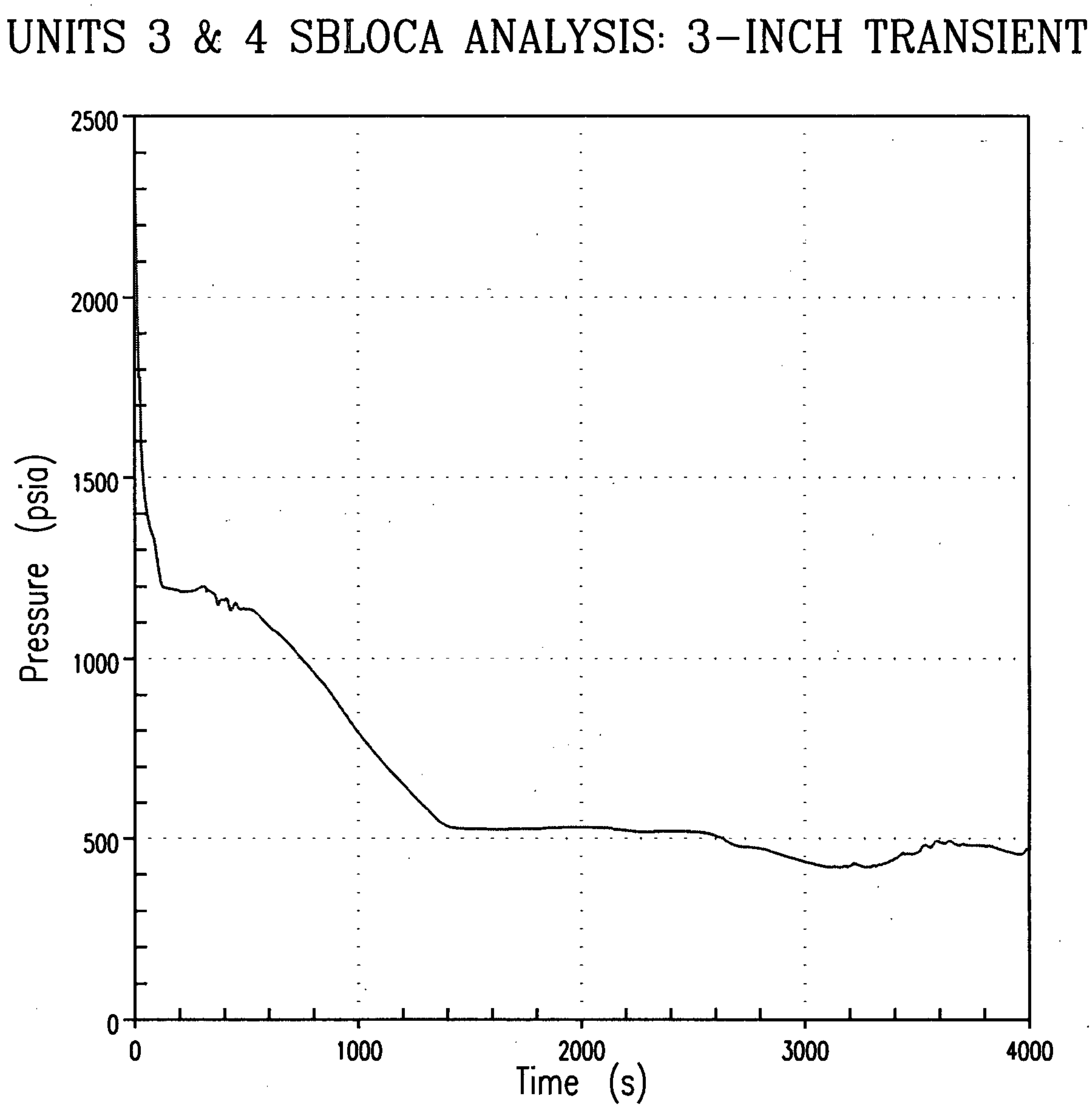


Figure 1.3.37-5: 3-Inch Break – Pressurizer Pressure

UNITS 3 & 4 SBLOCA ANALYSIS: 3-INCH TRANSIENT

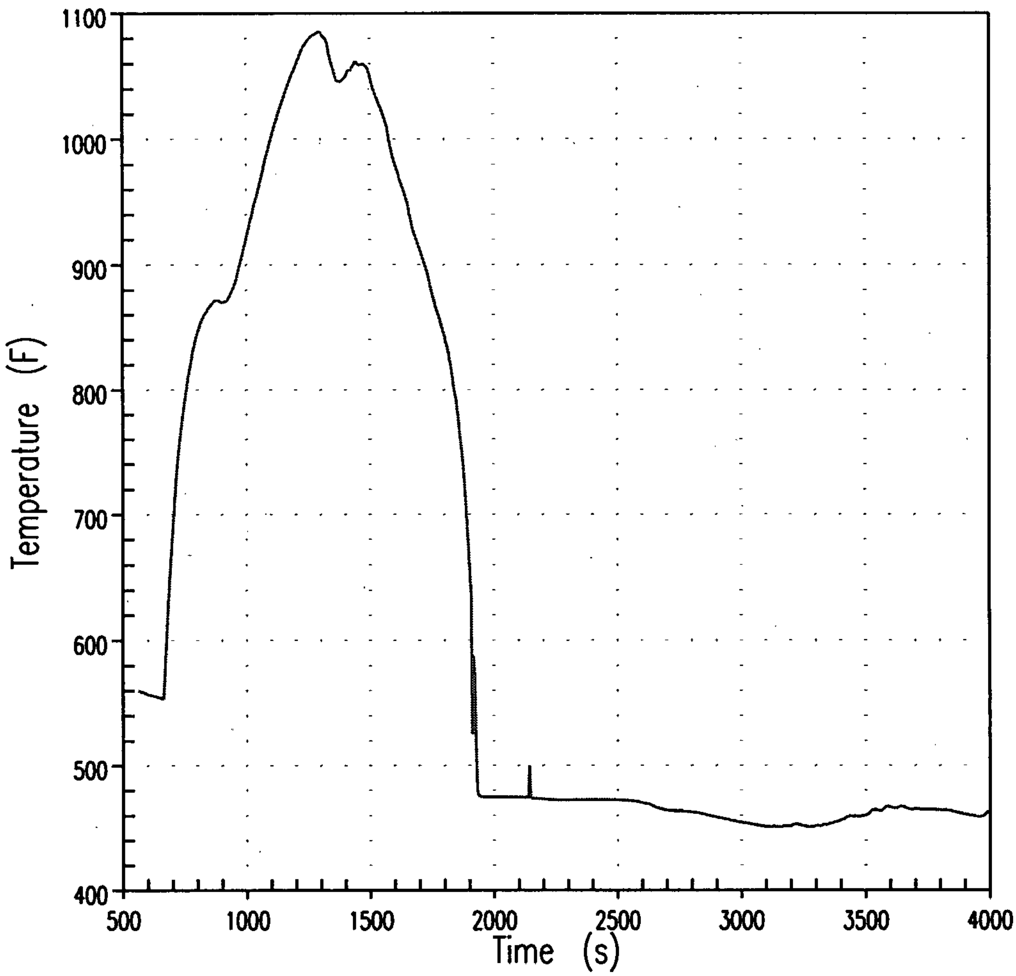


Figure 1.3.37-6: 3-Inch Break – Cladding Temperature at PCT Elevation

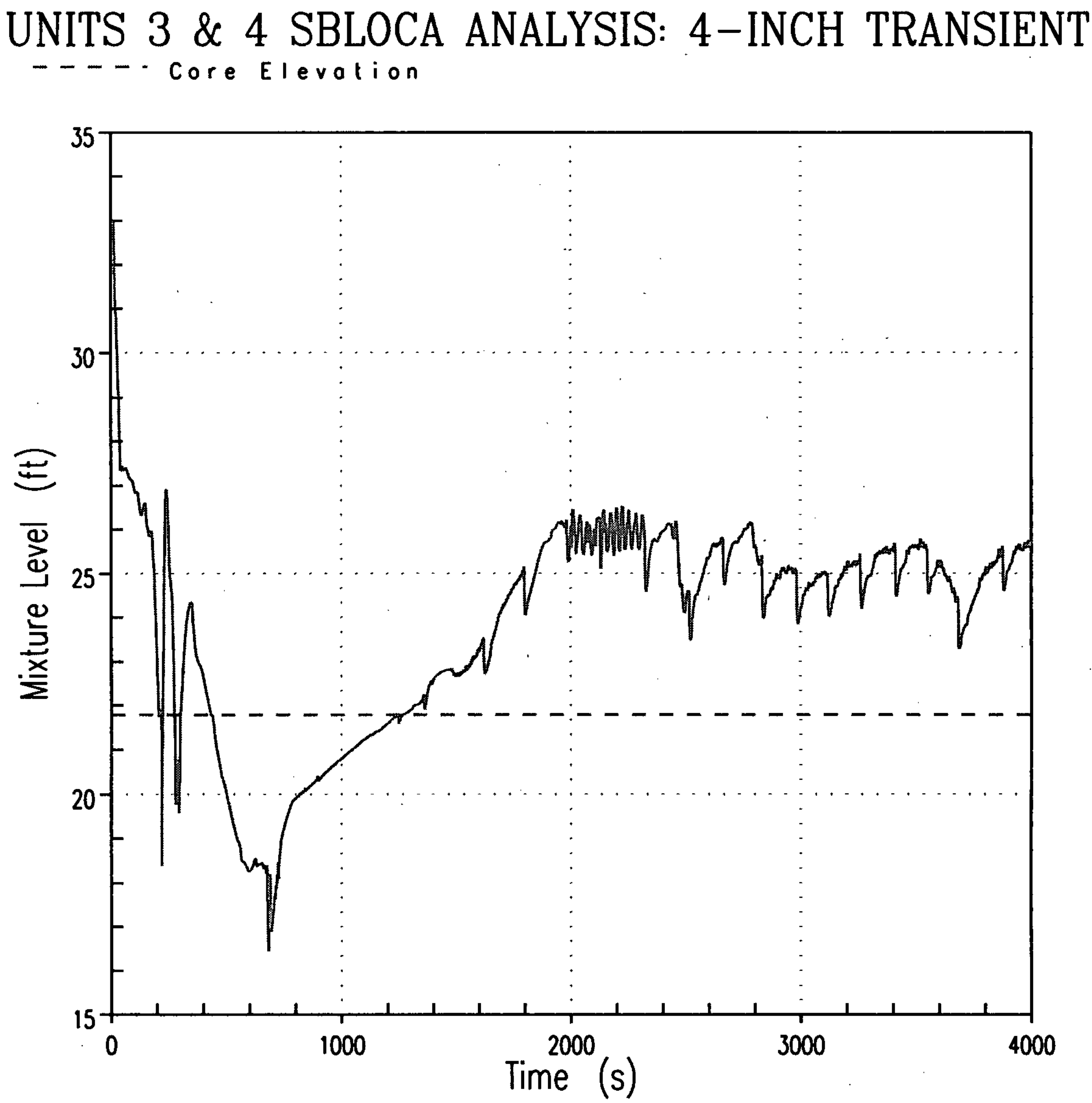


Figure 1.3.37-7: 4-Inch Break – Core Mixture Level

UNITS 3 & 4 SBLOCA ANALYSIS: 4-INCH TRANSIENT

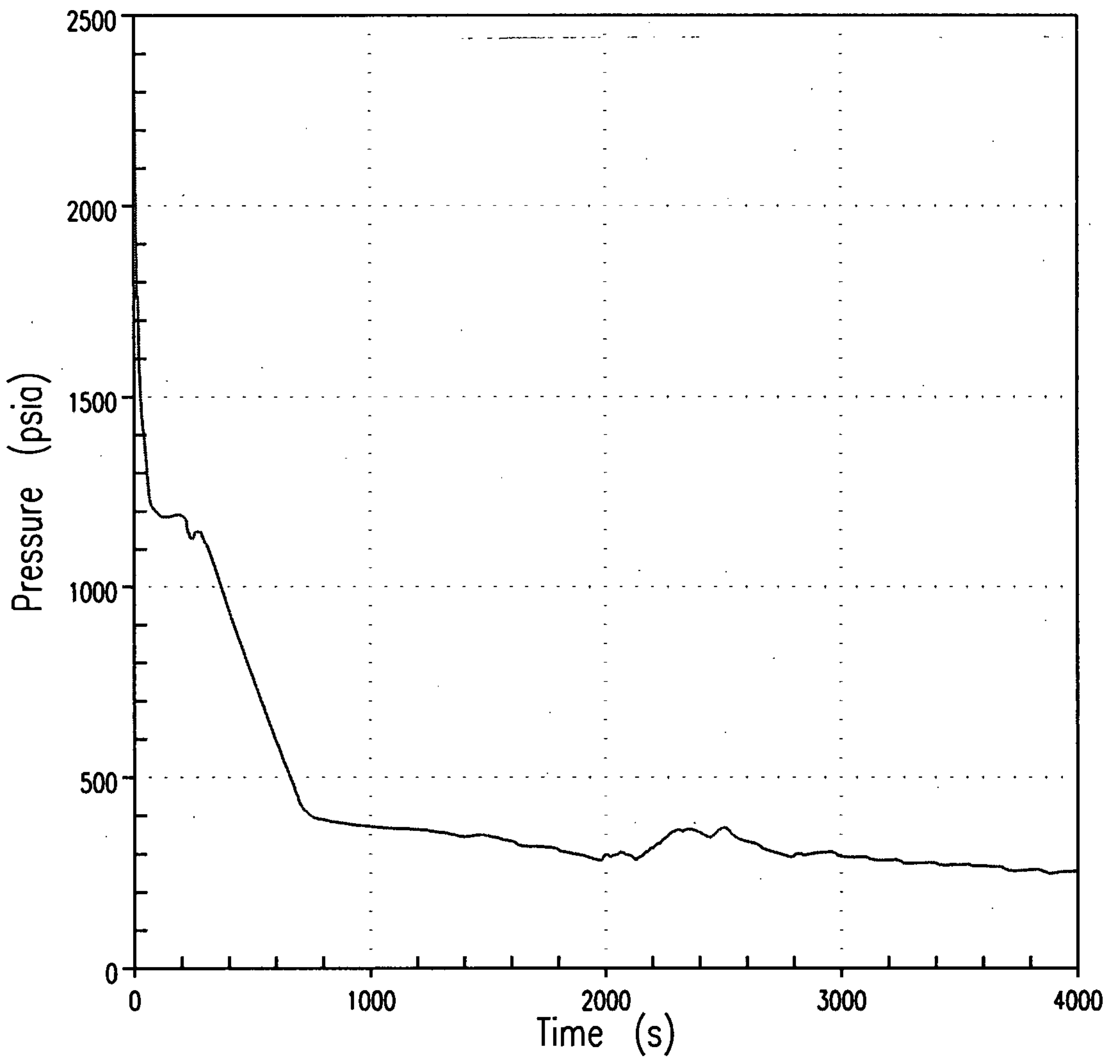


Figure 1.3.37-8: 4-Inch Break – Pressurizer Pressure

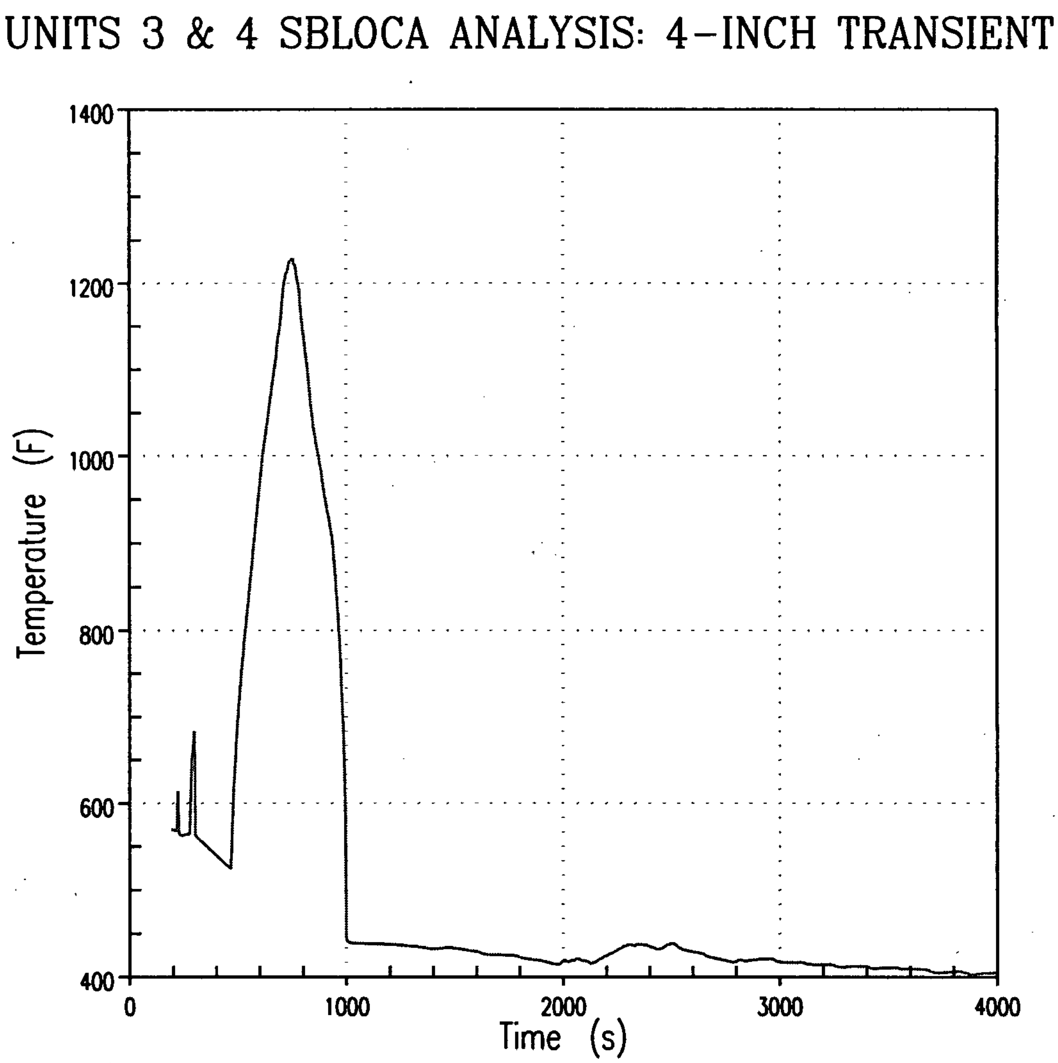


Figure 1.3.37-9: 4-Inch Break – Cladding Temperature at PCT Elevation

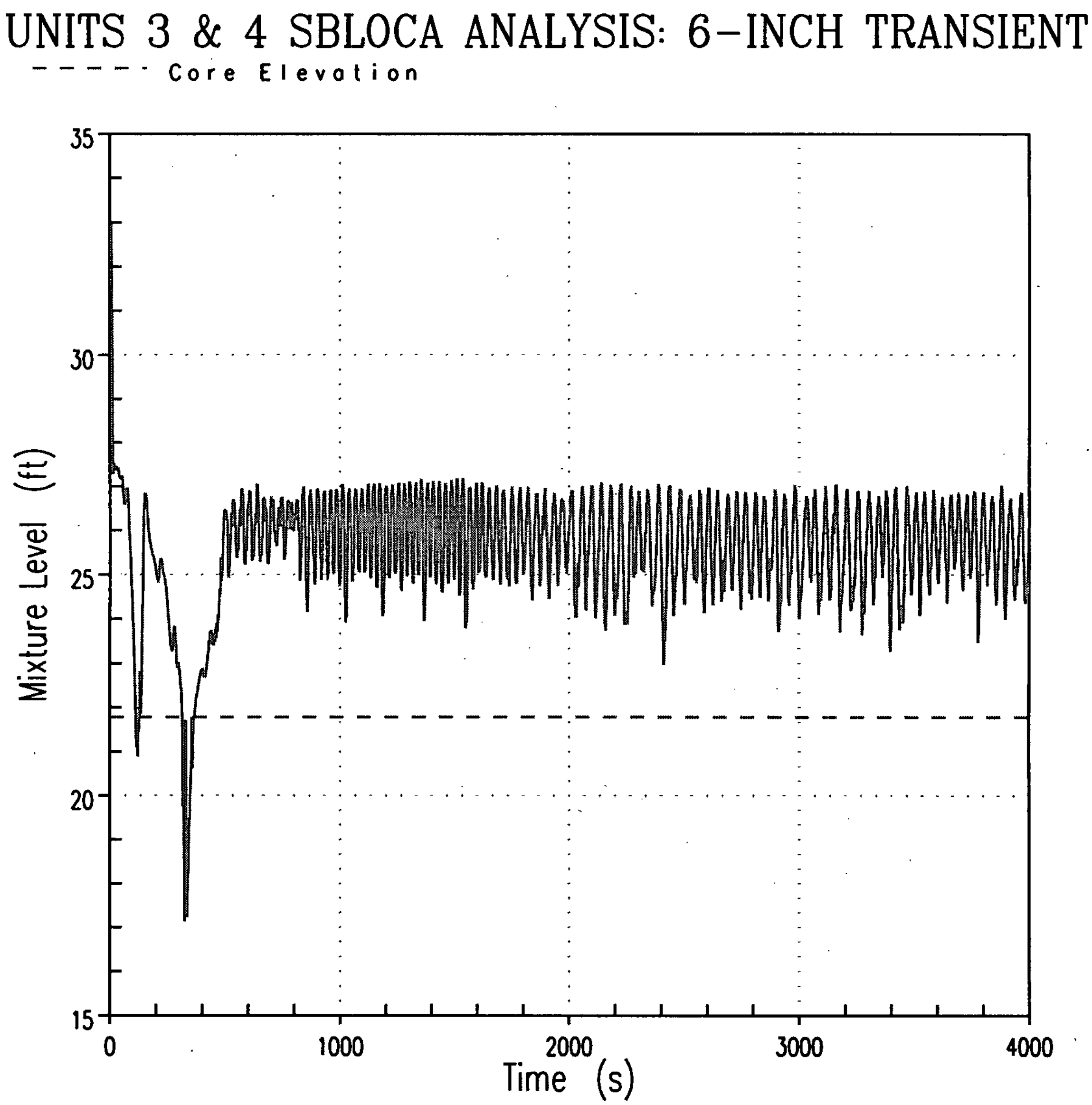


Figure 1.3.37-10: 6-Inch Break – Core Mixture Level

UNITS 3 & 4 SBLOCA ANALYSIS: 6-INCH TRANSIENT

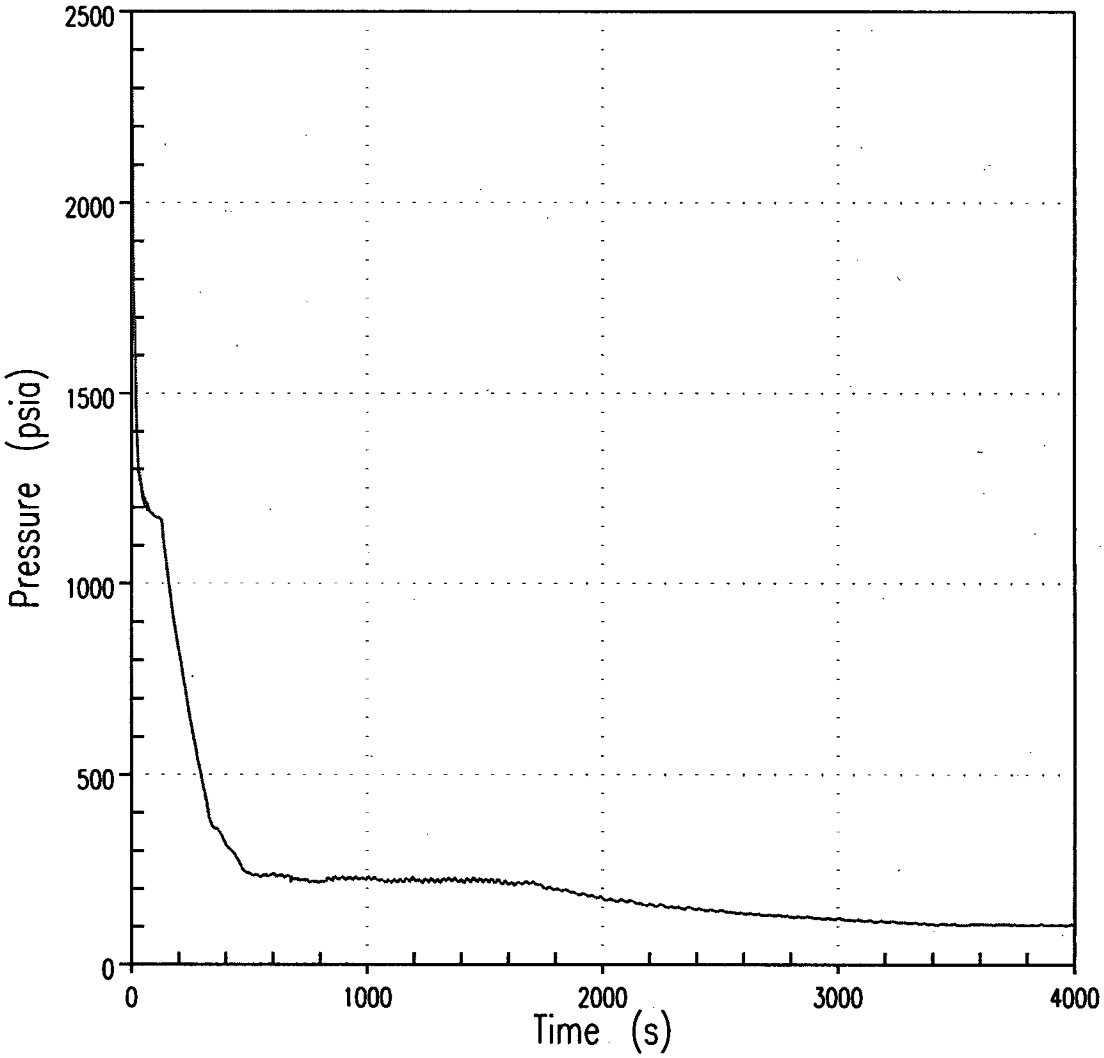


Figure 1.3.37-11: 6-Inch Break – Pressurizer Pressure

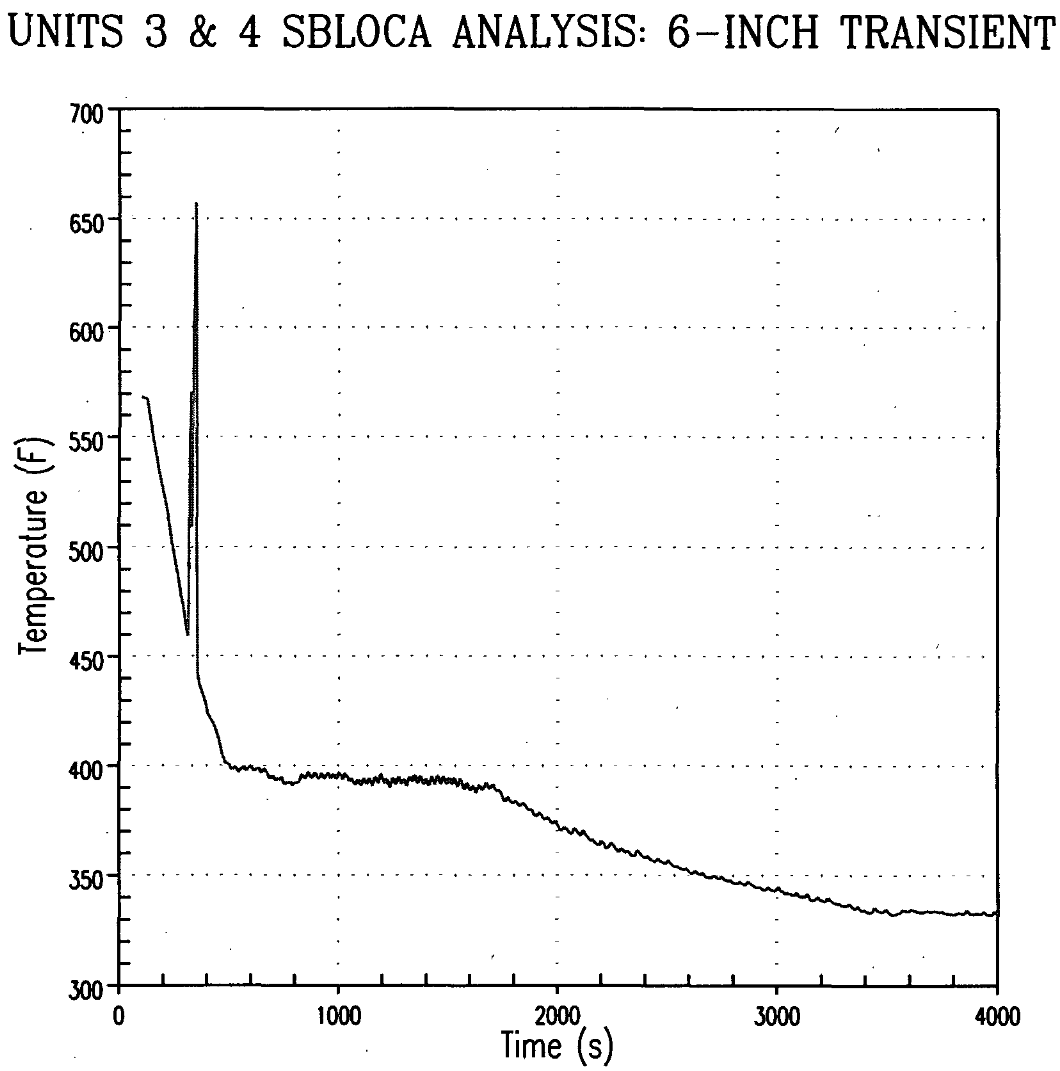


Figure 1.3.37-12: 6-Inch Break – Cladding Temperature at PCT Elevation

SRXB-1.3.38 10 CFR 50.46(a)(1)(i) requires that an acceptable emergency core cooling evaluation model be used to predict emergency core cooling behavior under a number of postulated loss of coolant accidents of sizes, locations, and other properties sufficient to provide assurance that the most severe loss of coolant accidents have been calculated. Although the coarse break spectrum does not explicitly address this requirement, a letter from Gresham, Westinghouse, to the NRC, dated July 2006, provides additional justification. It asserts that, provided the following:

a. The small-break PCT remains less than 1700°F, and
b. The large-break PCT significantly exceeds the small break PCT,

the analyzed break spectrum provides the requisite assurance. The letter includes sensitivity studies to demonstrate that the analysis of a finer break spectrum does not result in the prediction of a significantly higher PCT. Provide the applicability of this study to Turkey Point, specifically addressing each of the following:

a. How do the reactor coolant and emergency core cooling system designs at Turkey Point differ from those analyzed in Plant 1 (a three-loop, high-PCT plant)?

b. How do the differences identified in Item a), above, affect the emergency core cooling performance and its sensitivity to break size?

c. What key phenomena cause the small break peak cladding temperature to reach its maximum between 3 and 6 inches?

d. For the cluster of ASTRUM slot break results at 1200°F (PCT) vs. 0.5 (CD*Abreak/ACL), provide a table correlating the discharge coefficient (CD) and break area (Abreak) to the PCT. Explain why the ASTRUM results for the lower end of the range of analyzed break sizes are significantly different from the NOTRUMP results for the 6-inch cold leg and 8.75-inch safety injection line breaks.

Per Reference 9, the operating plants with NOTRUMP evaluation model (NOTRUMP-EM) analyses are categorized based on plant type (high peak cladding temperature (PCT) 3-Loop, high PCT 4-Loop, low PCT 4-Loop, and low PCT 2-Loop) due to overall similarities in their nuclear steam supply system (NSSS) designs and transient responses. A temperature of 1700°F was chosen as the designator between high and low PCT because of its margin to the 10 CFR 50.46 PCT limit of 2200°F and the increasing significance of the metal-water reaction, as predicted by the Baker-Just model, near this temperature.

The depressurization of the reactor coolant system (RCS) is driven by the rates at which steam is generated in the core, condensed within the reactor coolant system, and vented from the break for Licensing Basis small break loss-of-coolant accident (SBLOCA) calculations performed with the NOTRUMP-EM. The relative rates of RCS mass loss and system depressurization are dictated by the break size. The break size for which the limiting PCT is calculated to occur is a result of the emergency core cooling system (ECCS) response to RCS pressure. For the Turkey Point Units 3 and 4 EPU 3-inch break, the RCS depressurizes relatively slowly, reaching a pressure slightly below the accumulator gas cover pressure late in the transient; the accumulator flow remains relatively low such that the core recovers primarily on high head safety injection (HHSI). For the 6-inch break, the initial rate of RCS mass loss and depressurization is relatively high and the RCS quickly depressurizes which results in the accumulators injecting a large amount of inventory early in the transient recovering the core prior to any significant fuel rod heat-up. The 4-inch break experiences rates of mass loss and depressurization greater than those of the 3-inch break and less than those of the 6-inch break. The accumulators inject at an earlier time and higher flow rate than in the case of the 3-inch break; however, this is not sufficient to recover the core prior to experiencing fuel-rod heat-up. Due to the competing effects of vessel depletion and HHSI / accumulator delivery with increasing break size, a break size exists for which the depth and duration of core uncovery lead to the limiting cladding heat-up; for Turkey Point Units 3 and 4 at EPU conditions, this break size occurs between 3 and 6 inches, and is well represented by the 4-inch break.

There are several differences between the individual plants represented by the generic 3-loop plant presented in Reference 9, including fuel array size, vessel internals configuration (e.g., barrel-baffle region flow configuration (upflow vs. downflow) and thermal shield vs. neutron panels), and steam generator model; however, the Turkey Point Units 3 and 4 initial RCS inventory, core thermal power, and general RCS layout are comparable to the generic 3-Loop plant presented in Reference 9.

One unique aspect of the physical layout of the Turkey Point ECCS is the HHSI/Accumulator interaction (shared cold leg connection for the Accumulators, HHSI, and RHR). SBLOCA analyses typically include a severed ECCS line which results in the faulted loop SI flow spilling into containment. Since all ECCS flow is delivered via the accumulator line for Turkey Point Units 3 and 4, the 8.75-inch accumulator line break represents a severed SI line. The SBLOCA analyses for most 3-Loop plants consider a complete severing of a HHSI line which typically is a smaller diameter line. Due to the relatively large size of these breaks (severed ECCS lines), the RCS depressurizes quickly and results in core recovery via the available accumulators if core uncovery occurs. Severed ECCS line breaks are typically found to be non-limiting; therefore, this design feature does not impact the sensitivity of Turkey Point Units 3 and 4 to the refined break spectrum discussed in Reference 9.

The Turkey Point ECCS is generally similar to Plant A of Reference 9 in that each train of active ECCS equipment includes a HHSI pump and low head safety injection capability that is provided by a residual heat removal (RHR) pump. The capacities of the HHSI and RHR pumps are comparable to that of the generic plant evaluated in Reference 9 due to the minimum design requirements issued by Westinghouse for the pumps. Plant specific variations in accumulator volume and cover gas pressure and ECCS piping resistances will influence the transient response, but not to the degree that the conclusions of

Reference 9 are invalidated. The most unique aspect of the Turkey Point ECCS system relative to other 3-Loop plants, represented in Reference 9 by Plant A, is the sharing of ECCS equipment between Units 3 and 4. In the event that a safety injection signal (S-signal) is generated, all available HHSI pumps (2 redundant trains from each unit) will align and inject into the unit for which the S-signal was generated.

The SBLOCA analysis credits one HHSI pump from the affected unit and one HHSI pump from the unaffected sister unit injecting into the RCS. The analysis assumes that both operating HHSI pumps draw ECCS water from the refueling water storage tank (RWST) of the affected unit from the initiation of the transient in order to minimize the time required to draindown the RWST and initiate the transfer to cold leg recirculation. Without credit being taken for this capability the Turkey Point Units 3 and 4 transient behavior would be well represented by Plant A of Reference 9. The results of the SBLOCA EPU analysis (presented in the Response to RAI SRXB-1.3.37, Figures 1.3.37-1 – 1.3.37-12) show that the additional flow from a second train of HHSI reduces the amount of time required for the make-up flow to exceed the break flow, recover the core, and terminate the cladding heat-up portion of the transient compared to the 3-Loop high PCT plant of Reference 9.

Due to their lower capacity ECCS, the high PCT 3-Loop and 4-Loop (Plants A and B, respectively) plants are more sensitive to changes in RCS pressure and therefore tend to exhibit deeper and longer duration core uncovery for relatively small increases in RCS pressure. This sensitivity to RCS pressure results in the calculated PCT for these plants being more sensitive to the refined break spectrum than that of the higher ECCS capacity plants represented in Reference 9 by Plants C and D. Turkey Point Units 3 and 4 have ECCS injection flow available from two HHSI pumps that is comparable to the flow available (per MWt) for the low PCT 2-Loop and low PCT 4-Loop plants; therefore, the cladding temperature sensitivity to 0.25-inch break increments and associated changes in RCS pressure will have a minor effect, if any, on the results of the analysis as completed using the integer break spectrum.

The results of the best-estimate large-break LOCA (LBLOCA) slot breaks grouped near a 1200°F PCT (Reference 10, Figure 2.8.5.6.3.2-1) are summarized in Table 1.3.38-1. It can be seen from Table 1.3.38-1 that the flow area of the smallest breaks modeled in the ASTRUM analysis is near 1 ft^2. The 6 and 8.75-inch breaks modeled in the NOTRUMP small break analysis have less than half of the break flow area of the smallest analyzed LBLOCA transient. The difference in break area results in a significant impact on the RCS depressurization, depletion of mass from the vessel, and resulting core uncovery; therefore, the differences between the NOTRUMP and ASTRUM results are expected.

Table 1.3.38-1: Summary of Select ASTRUM and NOTRUMP Results

Evaluation Model	Break Size (NOTRUMP)/ Run Number (ASTRUM)	PCT (°F)	Break Flow Area (ft^2)	CD	Effective Break Area
NOTRUMP	6-inch	658	0.196	1.00	N/A
	8.75-inch	N/A	0.418	1.00	N/A
ASTRUM	6	1190	1.11	0.94	0.253
	106	1187	1.28	0.87	0.270
	11	1204	1.23	0.94	0.280
	119	1172	1.22	1.10	0.325
	27	1183	1.47	0.91	0.324
	46	1185	1.43	1.05	0.364
	14	1183	1.56	1.09	0.412
	60	1168	2.01	0.97	0.473
	96	999	1.85	1.11	0.498
	68	1243	2.12	1.00	0.514
	50	1058	2.33	0.93	0.525
	110	1037	2.39	0.91	0.527
	117	1124	2.20	1.12	0.597

SRXB-1.3.39 Since NOTRUMP was approved, more data concerning loop seal clearing phenomena became available. Considering post-1986 experimental data concerning loop seal clearing, justify the adequacy of the loop seal clearing modeling approach and attendant two-phase level depression and cladding heatup.

During the development of the NOTRUMP SBLOCA evaluation model (EM), several aspects of the reactor coolant system (RCS) required additional attention because of its influence on transient response, especially with regard to break location. One area in particular is the reactor coolant pump suction cross-over leg (loop seal). In development of an appropriate loop seal model, a test program, consisting of a 1/3 linear scale air-water mock-up of a Westinghouse PWR loop seal [Reference 11]), was conducted to investigate the loop seal clearing phenomena. This separate effects test data was then used to investigate key phenomena, such as vertical and horizontal entrainment mechanisms, in order to appropriately predict the residual liquid mass in the U-bend of this area of the RCS loop piping. Ultimately, this lead to the development and validation of the detailed NOTRUMP loop seal model as presented in Section 3-2-9 of WCAP-

10054-P-A [Reference 12]. As described in Section 6-1-1 of Reference 12, a simplified loop seal model was developed by changing the fluid node and flow link representations. This was done in an effort to save on computer resource time that was of consideration when the NOTRUMP-EM was first developed. The simplified loop seal model was benchmarked against the detailed model over a range of small break LOCA analyses and break sizes. The results of this benchmarking found that the simplified loop seal model showed comparable, conservative, transient results to the detailed model, and that it was suitable for predicting the loop seal clearing behavior for small break LOCA analyses. As such, the simplified model became the basis for the RCP suction cross-over leg model for the NOTRUMP SBLOCA EM and has remained so since that time.

While additional data has become available with regard to loop seal clearing since the development of NOTRUMP, most notably from ROSA, UPTF, and VVER-1000 (References 13, 14 and 15), the conclusions of these tests did not yield any new significant aspects that were not already known from previous tests such as SEMI-SCALE (See Section 7-3-1 of Reference 12) and the Westinghouse air-water tests (Reference 11). That is, for cold leg breaks, there will be a period where vapor generated in the reactor core does not have a significant vent path to the fault. However, once the liquid phase inventory in a stratified RCS drains down to the break elevation, periodic or sustained venting of steam through the loop seal will commence. As discussed below, NOTRUMP is considered to adequately possess the capabilities needed to demonstrate this with respect to a systems code that was developed to Appendix K standards.

The passage of vapor phase through liquid phase trapped in the loop seal can become an extremely complicated set of transitional flow regimes which can vary from slug to wavy-stratified to stratified. All of these will have dependencies on loop seal geometry, core steaming rates, local pressure, ECCS flow, RCP and steam generator status, etc. While NOTRUMP lacks a detailed horizontal flow regime map and transition criteria to capture all of the flow regimes that could occur in the loop seal, the model that exists is considered adequate to capture core mixture level depression, downcomer mixture level and related break quality changes that are characteristic in the RCS before and after the loop seal(s) clear. The fuel rod heat-up noted during loop seal depression is a momentary event with the core re-quenching after initial clearing. As such, the heat-up itself is not necessarily a first order impact other than the steam generation it brings after the core re-quenches. This steam burst can help clear other loops seals or further clear the purged loop seal depending again on the variables listed above. The main consideration is the amount of time the downcomer mixture level stays elevated with respect to the bottom of the cold leg while the core level is depressed prior to loop seal clearing since the RCS mass loss for a given pressure will be much greater due to low break quality (back-flow from the downcomer region). In addition, the RCS cannot begin to depressurize on its own accord until a sustained vent path for core vapor generation can be established. Therefore, the time at which the loop seal clears becomes an important factor. This time is mainly governed by the drain-down of the RCS (absent the realistic aspects discussed above) to where the water level in the SG outlet piping is approximately even with the top of the horizontal run of the loop seal piping. This is illustrated below in Figures 1.3.39-1 through 1.3.39-3. Once this liquid-vapor plane is established at the horizontal elevation, vapor flow can proceed through this area. The NOTRUMP model, which is drift flux based, shows that this vapor flow increases rapidly but that an oscillatory affect takes place for one period before extended venting is established and the pressure difference between the downcomer and upper plenum is relieved. Realistically, this oscillation could be considered a Kelvin-Helmholtz wave instability in the horizontal section of the loop seal. However, because of the limited code capability and models in NOTRUMP, this really is not the case. In the NOTRUMP simulations the noted effect (the temporary venting reduction) is considered more of a momentary increased vapor phase pressure drop not only in the simplified loop seal model, but also in other areas of the faulted loop due to liquid hold up before and during loop seal clearing. As such, while NOTRUMP may not capture this in a true phenomenological sense, the loop seal clearing behavior is considered representative. This can be further concluded when reviewing key aspects (such as downcomer-upper plenum differential pressure and loop seal water level response) presented in the Reference 14 UPTF report with respect to NOTRUMP loop seal clearing behavior.

During a small break LOCA there realistically may be a period of time when more than one loop seal will vent steam. Vapor venting through an intact loop in conjunction with the broken loop will result in beneficial steam condensation and mass redistribution, thus resulting in less severe core uncovery. In general, there is a threshold break size where more than one loop seal will vent steam for an extended period of time. Above this threshold, multiple loop seals may sustain venting (Reference 16). Below the threshold, sustained venting can only be expected to occur in one loop seal. As such, an artificial loop seal restriction is applied to the NOTRUMP model for breaks less than the threshold break size, only allowing the faulted loop's loop seal to clear. This artificial restriction is applied for breaks less than [ ]^{a,c} even though venting of multiple loop seals may be possible. This is considered to over penalize the predicted transient response because of the increased faulted loop pressure drop for break sizes in that break range (including the typical limiting break size; 4-inch break for the Turkey Point EPU). As a result, core level depression resulting from loop seal plugging is maximized.

Based on the loop seal model development, validation, and artificially imposed clearing restriction, the NOTRUMP-EM loop seal model is deemed conservative with respect to the more recent loop seal clearing experimental data (References 13, 14, and 15) and in compliance with 10 CFR 50 Appendix K.

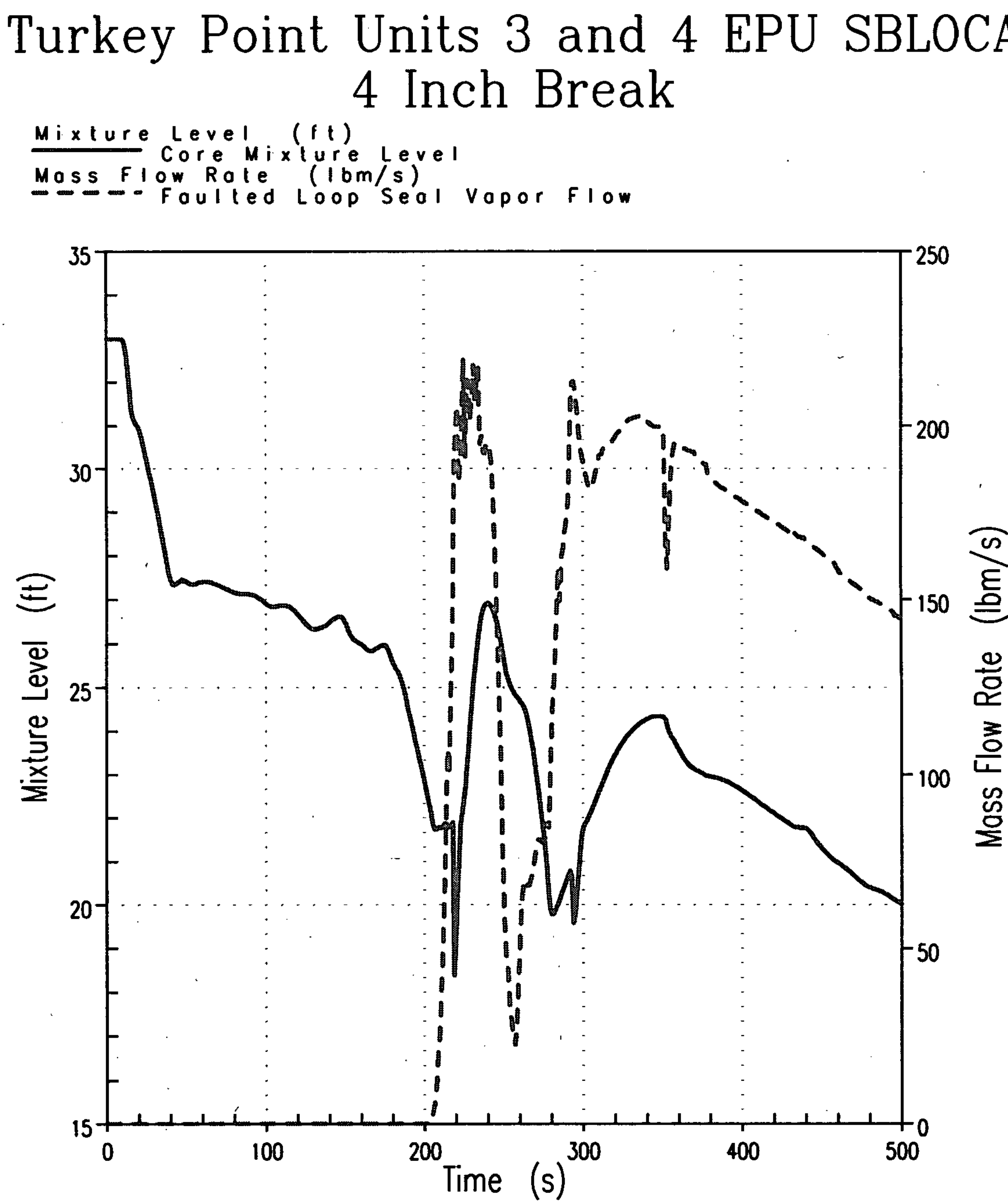


Figure 1.3.39-1

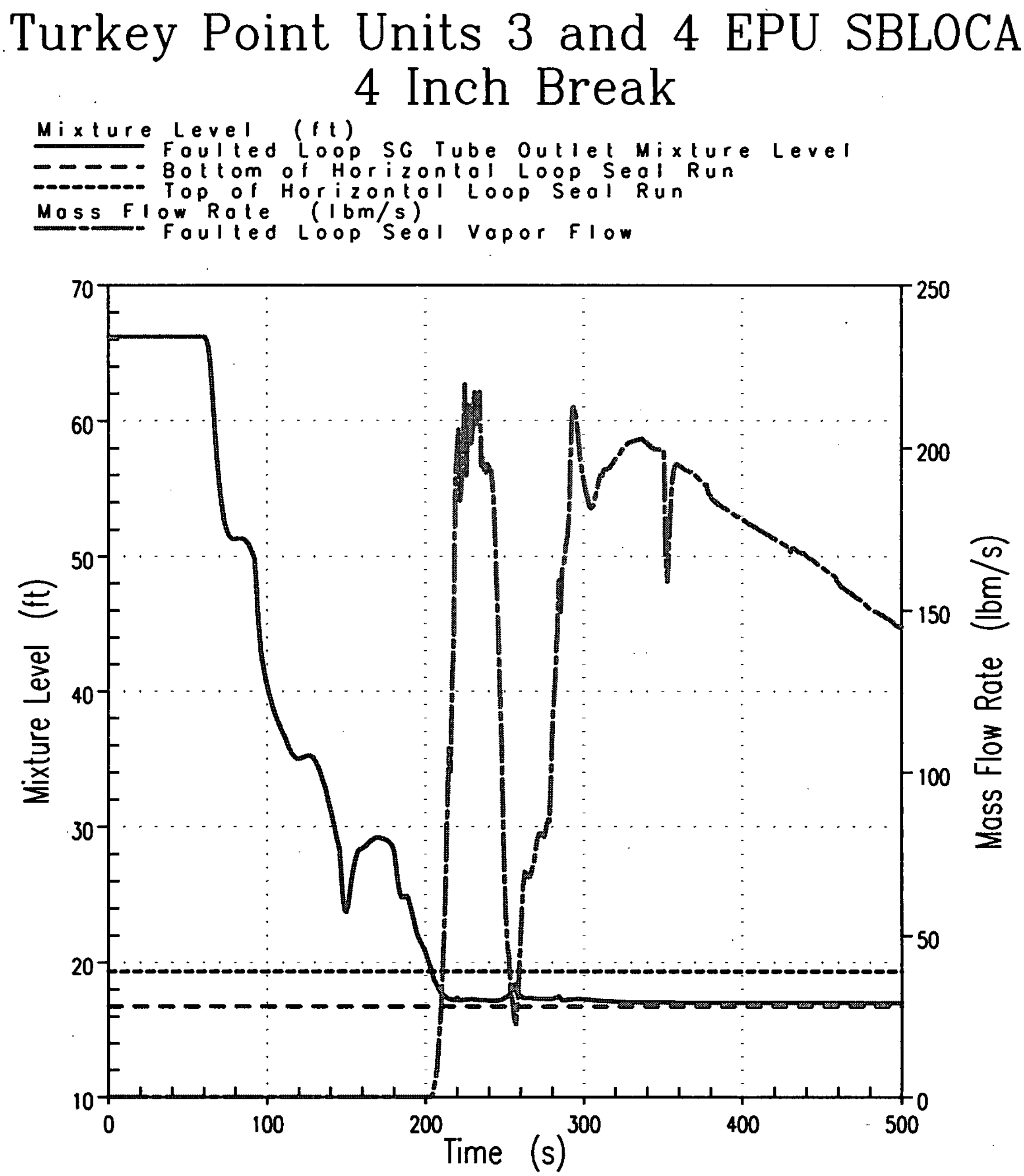


Figure 1.3.39-2

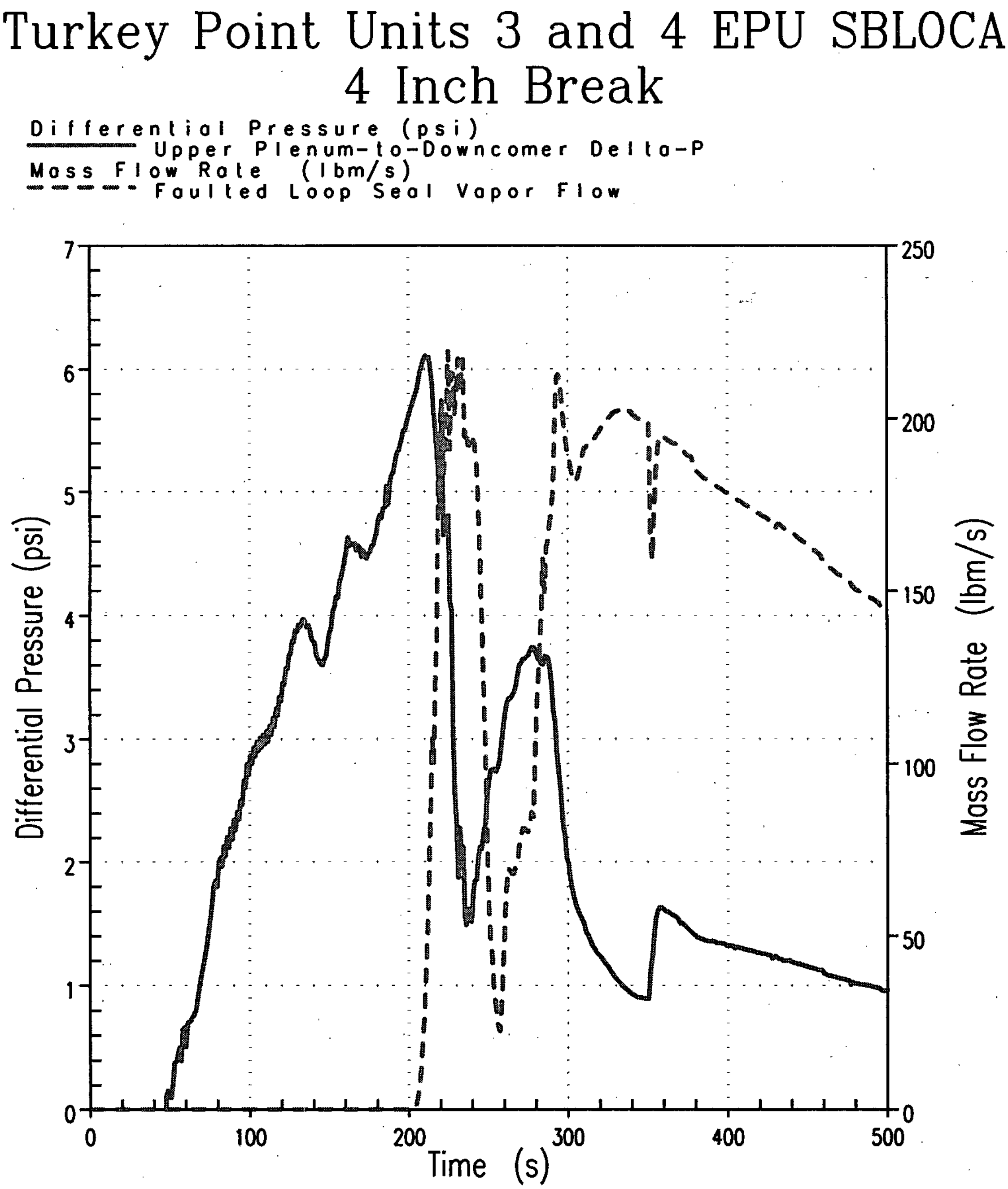


Figure 1.3-39-3

References

1. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request for Extended Power Uprate (LAR 205)," Accession No. ML103560169, October 21, 2010.
2. Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU – Reactor Systems (SRXB) Request for Additional Information - Round 1.3 (Part 3)", July 21, 2011.
3. Standard Review Plan, NUREG-0800 (formerly NUREG-75/087), Rev 3, March, 2007
4. Point Beach Nuclear Plant, Units 1 and 2 – Safety Evaluation Regarding Extended Power Uprate (TAC Nos. ME1044 and ME1045), Accession No. ML110450159, May 3, 2011.
5. W. J. Johnson (Westinghouse) to R. C. Jones (NRC)(NS-NRC-89-3466), October 23, 1989.
6. Appendix E of the Westinghouse testimony in the matter of Acceptance Criteria of Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors (Docket No. RM50-1).
7. WCAP-14565-P-A "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.
8. Bajorek, S. M., et al., March 1998, "Code Qualification Document for Best Estimate LOCA Analysis," Volume 1 Revision 2, and Volumes 2 through 5, Revision 1, WCAP-12945-P-A (Proprietary).
9. "Transmittal of LTR-NRC-06-44 NP-Attachment, 'Response to NRC Request for Additional Information on the Analyzed Break Spectrum for the Small Break Loss of Coolant Accident (SBLOCA) NOTRUMP Evaluation Model (NOTRUMP EM), Revision 1', (Non-Proprietary)," Accession No. ML062000027, July 14, 2006.
10. "Turkey Point Units 3 and 4 – License Amendment Request for Extended Power Uprate, Attachment 4; Licensing Report," Accession No. ML103560177, December 14, 2010.
11. Skwarek, R. J., "Experimental Evaluation of PWR Loop Seal Behavior During Small LOCA's," in ANS Specialists Meeting On Small Break Loss-Of-Coolant Analyses in LWR's, August 25-27, 1981, pp. 5-1 to 5-11, Monterey, CA (EPRI-WS-81-201).
12 WCAP-10054-P-A (Proprietary)/WCAP-10081-A (Non-Proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.
13. Osekaba et al., "Core Liquid Level Depression due to Manometric Effect during PWR Small Break LOCA," Journal of Nuclear Science and Technology, Feb 1987.
14. Liebert and Emmerling, "UPTF Experiment- Flow Phenomena during Full-Scale Loop Seal Clearing of a PWR," Nuclear Engineering and Design, Issue 179 1998.
15. Tuomisto and Kajanto, "Two-Phase Flow in a Full-Scale Loop Seal Facility," Nuclear Engineering and Design, Issue 107 1988.
16. N. Lee, et. al., ASME Paper 83-HT-104, "Phenomenological Uncertainty During Loop Seal Steam Venting in a Small Break Cold Leg LOCA of a PWR", July 1983.