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September 15, 1977

RELATED CORRESPONDENCE

DOCKETED USNRC
SEP 23 1977
Office of the Secretary
Docketing & Service
Section

Mr. John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Re: Docket No. 50-275-OL
Docket No. 50-323-OL
Diablo Canyon Units 1 and 2

Dear Mr. Stolz:

Your letter of June 1, 1977 requested additional information on the consequences of a postulated main steam line break accident inside containment. Answers to the individual questions in your request are provided in the attachment.

Very truly yours,

Philip A. Crane, Jr.

Attachments (10 copies)
CC w/attachment: Service List

REPLY TO NRC LETTER OF JUNE 1, 1977
CONCERNING ANALYSIS OF POSTULATED MAIN STEAM LINE BREAK ACCIDENT

A. Q. Provide single active failure analyses which specifically identify those safety grade systems and components relied upon to limit the mass and energy release and containment pressure/temperature response. The single failure analysis should include, but not necessarily be limited to: main steam and connected systems isolation; feedwater, auxiliary feedwater, and connected systems isolation; feedwater, condensate, and auxiliary feedwater pump trip, and auxiliary feedwater run-out control system; the loss of or availability of offsite power; diesel failure when loss of offsite power is evaluated; and partial loss of containment cooling systems.

A. Section 6.2.1 of the Diablo Canyon FSAR discusses the safety grade systems and components which are available to limit mass and energy releases to containment following a steam line rupture. The analysis presented in Section 6.2.1 of the FSAR has assumed that the steam line rupture is coincident with a single failure of either one emergency diesel generator unit, a steam line check valve, or a feedline control valve. The effect of each of these single failures is as follows:

1. Failure of one emergency diesel generator unit would result in the loss of one containment safeguards train, resulting in minimum containment heat removal capability.

2. Failure of a steam line check valve would allow reverse steam flow from the intact steam generators. The reverse flow is terminated by closure of the main steam line isolation valves in each of the steam lines. Also, a steam line check valve failure would increase the volume of steam which is not isolated from the break. When the check valve in the faulted line operates, the main steam piping volume capable of blowing down is located between the steam generator and the check valve. If the check valve fails, the volume between the break and the main steam line isolation valves in the other steam lines, including safety and relief valve headers and other connecting lines, will feed the break.

   Failure of a steam line check valve is more limiting than the failure of any single main steam line isolation valve (MSIV). A MSIV failure, regard-

less of location, would not result in reverse flow from the intact steam generators, as reverse flow would be prevented by closure of the check valve in the faulted steam line.

3. Failure of a feedwater control valve would result in additional feedwater being pumped into the steam generator associated with the broken steam line. Though the main feedwater pumps are tripped on any safety injection signal, the condensate system would continue to deliver condensate to the steam generator until the backup feedwater isolation valve (FIV) closed (maximum 60 second closure time).

   Failure of a feedwater control valve is more limiting than the failure of a FIV. A FIV failure would result in additional inventory in the feedline which would not be isolated from the steam generator. However, less feedwater would be pumped into the steam generator for a FIV failure, as main feedwater flow would be terminated by closure of the feedwater control valve (maximum 5 second closure time). The main feedwater pumps would also be tripped.

   For all cases analyzed in Section 6.2.1 of the Diablo Canyon FSAR, the auxiliary feedwater system was assumed to be operating at its peak capacity, with no credit taken for auxiliary feedwater run-out control systems.

B. Q. Discuss and justify the assumptions made regarding the time at which active containment heat removal systems become effective.

A. The active containment heat removal systems in the Diablo Canyon Nuclear Power Plant are the spray system and fan cooler system. The sprays and fan coolers are assumed to start 40 seconds after the containment transient begins.

C. Q. Discuss and justify the heat transfer correlation(s) (e.g., Tagami, Uchida) used to calculate the heat transfer from the containment atmosphere to the passive heat sinks, and provide a plot of the heat transfer coefficient versus time for the most severe steam line break accident analyzed.

A. A plot of the heat transfer coefficient versus time for the worst break ($1.4\ ft^2$ break with a failed steam line check valve) is given in Figure Q6.44-1. These heat transfer coefficients are based upon the work of Tagami (Reference 1). Figure Q6.44-1 compares the stagnant Tagami correlations (used for this analysis) with the Uchida correlation's heat transfer coefficients. This comparison illustates the conservatism of the Tagami coefficients used.

D. Q. Specify and justify the temperature used in the calculation of condensing heat transfer to the passive heat sinks; i.e., specify whether the saturation temperature corresponding to the partial pressure of the vapor, or the atmosphere temperature which may be superheated, was used.

A. The temperature used to calculate the heat transfer to the passive heat sinks is the containment atmosphere vapor bulk temperature.

E. Q. Discuss and justify the analytical model including the thermodynamic equations used to account for the removal of the condensed mass from the containment atmosphere due to condensing heat transfer to the passive heat sinks.

A. Complete revaporization of the condensate under superheat conditions is assumed by the COCO computer code. For a discussion and justification, see Reference 2.

F. Q. Provide a table of the peak values of containment atmosphere temperature and pressure for the spectrum of break areas and power levels analyzed.

A. Table Q6.44-1 provides a table of peak pressures for all breaks analyzed. Table Q6.44-2 provides peak temperatures for these breaks.

G. Q. For the case which results in the maximum containment atmosphere temperature, graphically show the containment atmosphere temperature, the containment liner temperature, and the containment concrete temperature as a function of time. Compare the calculated containment atmosphere temperature response to the temperature profile used in the environmental qualification program for those safety related instruments and mechanical components needed to mitigate the consequences of the assumed main steam line break and effect safety reactor shutdown.

A. Figure Q6.44-2 shows the containment atmosphere, liner, and concrete temperatures versus time for the worst case steam break.

A comparison of Diablo Canyon containment atmosphere response to the environmental qualification profile for transmitters shows that at about 10 seconds the containment temperature reaches 326°F which is 6°F above the envelope at that time. This peak exceeds the transmitter qualification envelope for a brief period of time (20 seconds), is a superheated condition (containment pressure is 22 psig), and occurs early in the transient. The instrument tests reported in Reference 5 demonstrate that this peak will not result in the equipment's temperature exceeding the qualification temperature. Comparing the containment atmosphere response to the qualification temperature profile for the Fan Cooler Motor (with Heat Exchanger) shows the containment response is 2°F above the peak temperature reached in the Fan Cooler Motor (with Heat Exchanger) test (324°F, 70 psig). However, for the same reasons as above, the peak will not result in the equipment's temperature exceeding the qualification temperature.

H. Q. For the case which results in maximum containment atmosphere pressure, graphically show the containment pressure as a function of time.

A. Figure Q6.44-3 shows the containment pressure versus time for the worst break.

I. Q. For the cases which result in the maximum containment atmosphere pressure and temperature, provide the mass and energy release data in tabular form.

A. Table 6.2-13A in the FSAR presents the mass and energy release rates for all of the breaks analyzed.

J. Q. For the instrumentation and equipment located inside the containment and required to (1) detect the steam line break; (2) initiate safety systems; and (3) monitor the course of the accident, provide the following:

1. A description of the tests which were/or will be performed to show that this instrumentation and equipment are/or will be qualified to perform their function before, during and after the accident. Include the spectrum of environmental conditions for which tests were/will be performed and state the accepetance criteria. The instrumentation and equipment to be considered includes; but is not limited to the following: (a) pressurizer

pressure and level sensors and transmitters; (b) steam generator pressure and level sensors and transmitters; (c) main steam line pressure, differential pressure and flow sensors and transmitters; (d) primary system hot leg and cold leg temperature sensors and transmitters; (e) primary system pressure sensors and transmitters; (g) feed water flow sensors and tramsitters; (h) containment pressure sensors and transmitters; (i) valve operators and position switches; (j) electrical cables, motors and penetrations; (k) containment coolers. Also identify any additonal instruments and equipment required.

2. A description of the separation and independence between redundant sensors, cables and other equipment associated with each steam generator and steam line.

3. A description of the independence and separation between each steam generator and between each steam line.

A. 1. The following instrumentation, located inside containment, is utilized to detect and initiate appropriate safety system functions in the event of a steam line depressurization of break (see Chapter 15 of FSAR).

a. Pressurizer Pressure and Level Transmitters (Condition II)

b. Steam Flow Transmitters and Reactor Coolant System RTD's (Narrow Range) (Condition III, IV).

The following instrumentation, located inside containment, is to be available for long term monitoring of the Condition (II, IV) accident. (see Chapter 7 of FSAR).

a. Pressurizer Level Transmitters

b. Narrow Range Steam Generator Water Level Transmitters

c. Wide Range RCS RTD's

d. Wide Range RCS Pressure

The installed Pressurizer Level Transmitters, Narrow Range Steam Generator Water Level Tranmsitters and Wide Range RCS Pressure Transmitter will be

supplied with instruments qualified as part of the Westinghouse Supplemental Program, Reference 3. The qualification conditions and acceptance criteria for these instruments are presented in Reference 4. The narrow range and wide range RCS temperature detectors, steam flow transmitters and pressurizer pressure transmitters have been qualified to the temperature/pressure conditions identified in Reference 4 for the length of time required by their functions (trip or long term monitoring).

The fan coolers located inside containment are required to function during a steam line break accident. Qualification testing of this equipment is described in WCAP 7829, "Fan Cooler Motor Unit Test." Requirements for this system are discussed in Chapter 6 of the Diablo Canyon FSAR.

Section 3.11 of the Diablo Canyon FSAR discusses the environmental qualification testing of wire and cable, electrical connections, and electrical penetrations. The qualification level temperatures for all equipment tested, with the exception of the containment fan cooler motors power cable and the electrical penetrations, are above the calculated peak containment atmosphere temperature for the postulated steam line break accident.

The power cable for the containment fan cooler motors was tested at temperatures up to 302°F at steam pressure of 50 psig for a period of approximately two hours. The calculated containment temperature response is above 302°F for approximately 15 seconds at the outset of the transient. Since the peak exceeds the qualification level for a breif period of time, is a superheated condition, and occurs early in the transent, the peak will not result in the cable temperature exceeding the qualification level temperature.

The electrical penetrations were successfully prototype tested by the manufacturer at a condition of 281°F, 63 psig, and 90-100% RH for a duration of 240 hours. While this qualification level temperature is below the short duration containment atmosphere temperature peak for a period of approximately 50 seconds, it is above the calculated temperature response of the containment liner and concrete. The electrical penetrations are essentially short sections of pipe embedded in the containment concrete. It is expected their temperature response would be bounded by the response of the containment liner and concrete.

2. All redundant sensors are mounted in separate panel enclosures. Tubing for redundant sensors has a minimum separation of 18 inches and is routed through separate openings. Cable from these sensors is run in separate conduits through separate penetrations. Conduits are separated by a minimum of one inch when crossing or adjacent to each other. Cables entering the protection racks from below are in separate cable trays isolated from each other by solid structural members. The individual channels of protection instruments are in separate racks. When more than one channel carries signals to the same control board, each channel is run in separate metal raceways.

3. The four Diablo Canyon steam generators are nearly symmetrical about the north-south and east-west centerlines of the containment building. Steam generators 1-1 and 1-2 are separated by 23'-5" center to center, as are steam generators 1-3 and 1-4. Steam generators 1-1 and 1-2 are essentially separated from 1-3 and 1-4 by the refueling canal; 58' center to center. The main steam and feedwater piping for steam generators 1-1 and 1-2 runs roughly parallel from the steam generators to the penetrations on the north side of the containment building. The main steam leads are 8'-9" apart, center to center, at the closest spacing inside containment. The feedwater piping is 18'-3" apart, center to center, at the closest spacing inside containment. The main steam and feedwater piping for steam generators 1-3 and 1-4 is opposite hand to the piping for 1-1 and 1-2.

REFERENCES

1. Tagami, Takasi, Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June, 1965 (No. 1)

2. Letter NS-CE-1183, of 8-27-76 from C. Eicheldinger of Westinghouse to J. F. Stolz of the NRC.

3. Letter NS-CE-692 from C. Eicheldinger of Westinghouse to D. Vassallo dated 7-10-75.

4. Letter NS-CE-1384 from C. Eicheldinger of Westinghouse to D. Vassallo of the NRC dated 3-23-77.

5. Hsieh, T., et al., "Environmental Qualification Instrument Transmitter Temperature Transient Analysis", WCAP 8936 (Proprietary), WCAP8937 (Non-proprietary), February 1977

TABLE Q6.44-1

CONTAINMENT PRESSURE RESPONSE TO STEAMLINE RUPTURES

	CASE	PEAK PRESSURE
1.	1.4 ft^2 Rupture with Diesel Generator Failure, Zero Power	40.54 PSIG
2.	1.4 ft^2 Rupture with Failed Feed Control Valve, Full Power	37.1 PSIG
3.	1.4 ft^2 Rupture with Failed Steam Line Check Valve, Zero Power	41.56 PSIG
4.	4.6 ft^2 Rupture with Failed Feed Control Valve, Full Power	33.01 PSIG

TABLE Q6.44-2

CONTAINMENT TEMPERATURE RESPONSE TO STEAMLINE RUPTURES

CASE	PEAK TEMPERATURE
1. 1.4 ft^2 Rupture with Diesel Generator Failure, Zero Power	250.12°F
2. 1.4 ft^2 Rupture with Failed Feed Control Valve, Full Power	274.93°F
3. 1.4 ft^2 Rupture with Failed Steam Line Check Valve, Zero Power	326.06°F
4. 4.6 ft^2 Rupture with Failed Feed Control Valve, Full Power	249.54°F

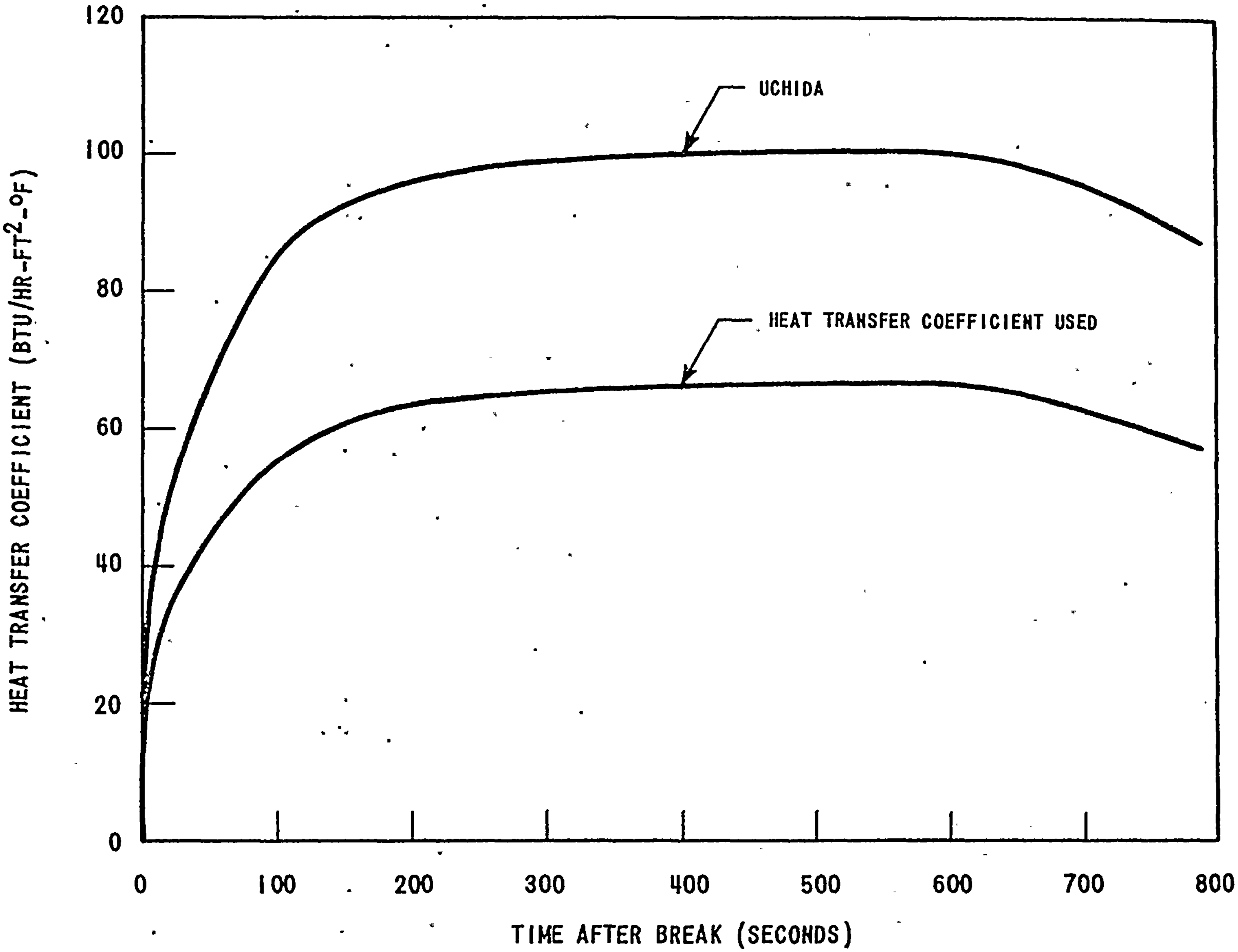

Figure Q 6.44 – 1

11,612-1

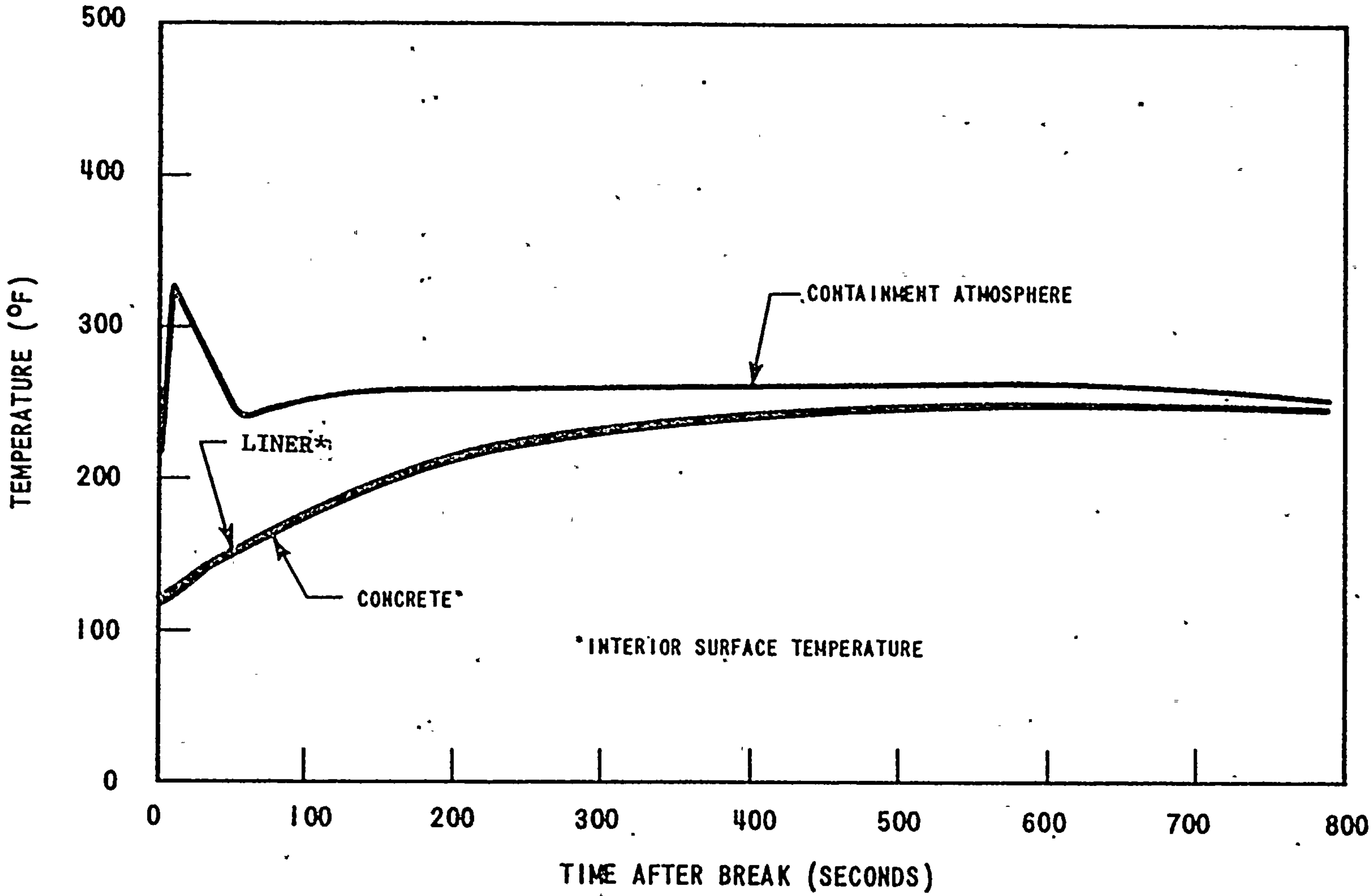

Figure Q 6.44 – 2

11,612-2

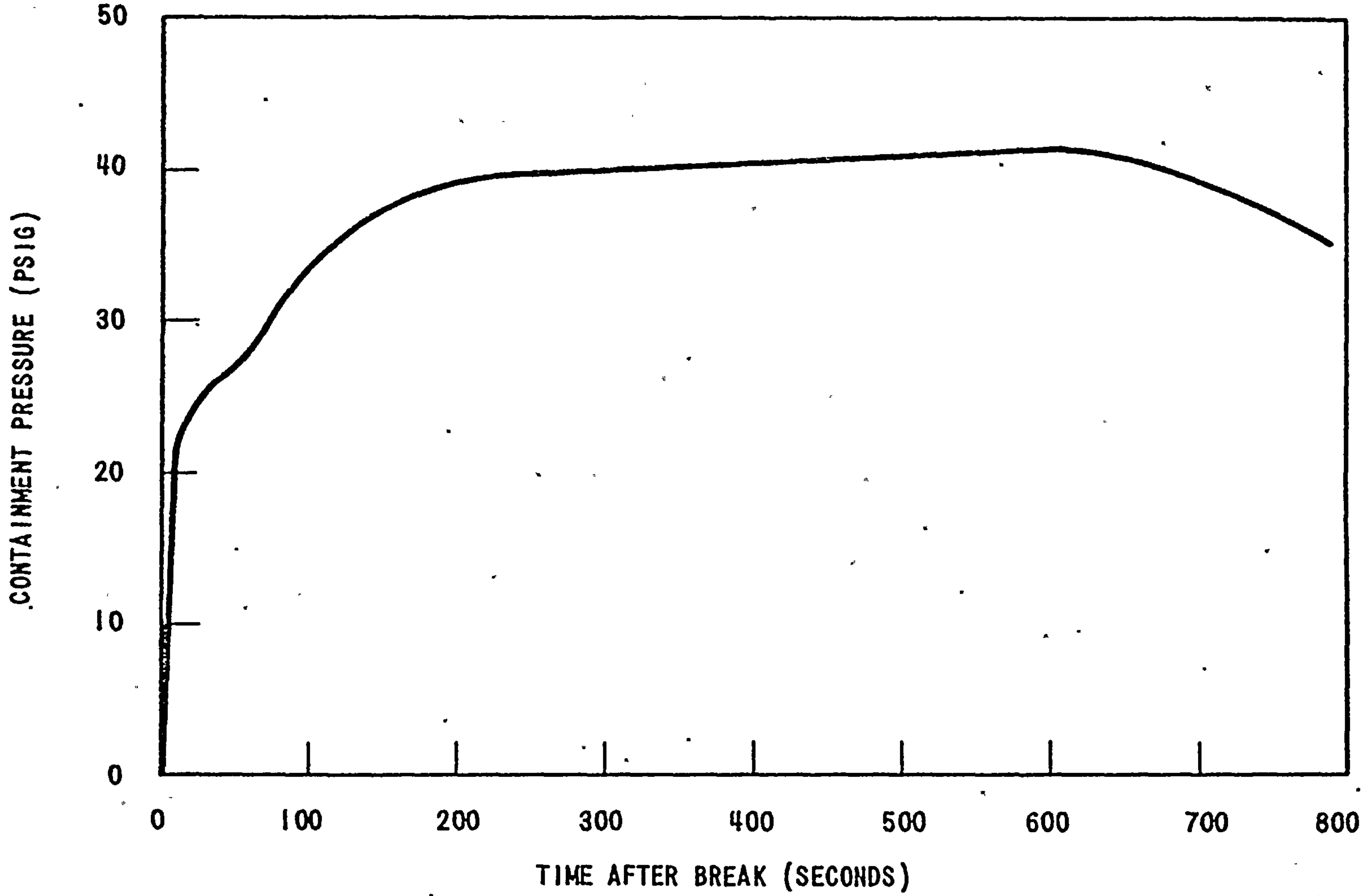

Figure Q 6.44 – 3

11,612-3