REVISION LOG SHEET

Revision Date: PORC 2-23-83 (issued 4-18-83)

This log sheet must be retained as the last page of the Browns Ferry Implementing Procedures Document.

Inserted by: ______________________ Date Inserted: ______________________

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List of Effective Pages	1 of 10	3/3/83	List of Effective Pages	1 of 10	3/11/83
	2 of 10	3/3/83		2 of 10	3/11/83

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT IMPLEMENTING PROCEDURES DOCUMENT

LIST OF EFFECTIVE PAGES

This List of Effective Pages must be retained with the Browns Ferry Nuclear Plant Implementing Procedures Documents.

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	IP-2		Coversheet	06/15/82
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ATTACHMENT 1

Contact as many people in each group as noted.

Message: "We have a/an (ALERT) (SITE AREA EMERGENCY) (GENERAL EMERGENCY) condition exsisting at the plant. Please report to the Operations Support Center immediately."

Initials	Time Contacted	Name	PAX	DIM	HOME
		INSTRUMENT MAINTENANCE (CONTACT 3)			
______	__________	Alton McCaleb			
______	__________	Gene Hartsfield			
______	__________	J. D. Thompson			
______	__________	Guy V. Thompson			
______	__________	Ken Montgomery			
______	__________	Ron Turberville			
		ELECTRICAL MAINTENANCE (CONTACT 4)			
______	__________	Jim Fowler			
______	__________	Mike Jackson			
______	__________	John Killen			
______	__________	Julian Bass			
______	__________	Pete McLemore			
______	__________	Billy Tompkins			
______	__________	Dennis White			

*Revision 25

REVISION LOG SHEET

Revision Date: PORC 3/25/83 (issued 4/20/83)

This log sheet must be retained as the last page of the Browns Ferry Nuclear Plant Implementing Procedures Document.

Inserted by: ____________________ Date Inserted: ______________

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Part	Page Number	Revision	Part	Page Number	Revision
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IP-6	1 of 4	02/14/83	IP-6	1 of 4	03/25/83
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TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT IMPLEMENTING PROCEDURES DOCUMENT

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NOTIFICATION OF UNUSUAL EVENT

1.0 PURPOSE

1.1 Provide for timely notification of appropriate individuals and organizations of a NOTIFICATION OF UNUSUAL EVENT.

1.2 Provide for periodic reanalysis to determine whether the NOTIFICATION OF UNUSUAL EVENT should be cancelled, continued or upgraded to a more serious classification.

2.0 INSTRUCTIONS

Initials Time

_______ ____ 2.1 Shift Engineer notify Operations Duty Specialist (40- 200, 50-0200,or 51-0200) within 5 minutes of declaration of NOTIFICATION OF UNUSUAL EVENT.

Give the following:

a. Your name.

b. Browns Ferry Nuclear Plant.

c. NOTIFICATION OF UNUSUAL EVENT.

d. Time of incident.

e. Brief description of incident.

f. Plant condition (whether stable or deteriorating).

g. Reactor (did/did not) shut down at (time).

h. Unusual release of radioactivity (yes, no or not known).

i. If radiation release: a) Ground Level, b) Elevated - Airborne, c) Waterborne, d) Other

j. Release rate if unusual release from Table 1 and 2 of IP-3 Release rate ________ μCi/sec.

k. Direction wind is coming from _______ (degrees) and speed_______(meters/second). (Use 300 ft. info. if available).

l. No protective action recommended.

m. Any emergency actions underway onsite.

n. Any offsite support that has been requested.

INITIALS TIME

INITIALS	TIME		
________	______	2.2	Operations Duty Specialist will return call to verify authenticity.
		2.3	Shift Engineer will notify the following of the event:
________	______		a. Other Shift Engineer. (When assigned)
________	______		b. STA (PAX
________	______		c. Operations Section Supervisor — R. Hunkapillar, Decatur
			OR
________	______		Operations Supervisor — Tommy Jordan, Muscle Shoals
			OR
________	______		Operations Supervisor — A. Burnette, Florence
________	______		d. Plant Superintendent — J. A. Coffey, Decatur
			OR
			Assistant Plant Superintendent — J. E. Swindell, Decatur
			OR
			Assistant Plant Superintendent — J. R. Pittman, Decatur
________	______		e. Public Information Officer — R. C. Boyer, Decatur, Beeper
________	______	2.4	Shift Engineer w'll notify the NRC of <u>NOTIFICATION OF UNUSUAL EVENT</u> by red phone. Give a brief description. Maintain an open line until released by NRC.

*Revision NJ

ATTACHMENT 1

Contact one or more AS NOTED.

MESSAGE: "We have a/an (ALERT) (SITE EMERGENCY) (GENERAL EMERGENCY) condition existing at the plant. This is not a drill. Please report to the Technical Support Center immediately as the (Job Title)."

SITE EMERGENCY DIRECTOR (CONTACT 1)

INITIAL	TIME CONTACTED		PAX	DIM	HOME
* ______	______	Jim Coffey			
______	______	Jim Swindell			
______	______	John Pittman			
______	______	Ray Hunkapillar			

OPERATIONS MANAGER (CONTACT 1)

______	______	Ray Hunkapillar
______	______	Tommy Jordan
______	______	A. L. Burnette

TECHNICAL ASSESSMENT MANAGER (CONTACT 1)

______	______	J. E. Swindell
______	______	W. C. Thomison
______	______	Dwight Mims

MAINTENANCE MANAGER (CONTACT 1)

______	______	John Pittman
______	______	John Miller
______	______	Tom Cosby

REP COMMUNICATOR (CONTACT 1)

______	______	Terry Chinn
______	______	Bill Roberts
______	______	Carroll Rozear

*Revision rN

SECRETARY (CONTACT 3)

INITIALS	TIME CONTACTED		PAX	DIM	HOME
		Glenda Harrison			
		Betty Riley			
		Cathy McChristian			
		Jane Page			
*		Sandra Strickland			
		Jacque Garner			

TSC COMMUNICATOR (CONTACT 1)

INITIALS	TIME CONTACTED	
		Bob Metke
		Dwight Mims
		Roger McPherson

NRC COMMUNICATOR (CONTACT 1)

INITIALS	TIME CONTACTED	
		Bill Roberts
		Terry Chinn
		Carroll Rozear

OPERATIONS SPECIALIST (CONTACT 1)

INITIALS	TIME CONTACTED	
		S. Burnette
		Tommy Jordan
		Roy Smallwood

*Revision M

HEALTH PHYSICS (CONTACT 2)

INITIAL	TIME CONTACTED		PAX	DIM	HOME
		Allen Sorrell			
		Ed Cargill			
		Wayne Simpkins			
		Herman Crowson			

RADIOCHEMICAL ENGINEER (CONTACT 1)

INITIAL	TIME CONTACTED		PAX	DIM	HOME
		Bill Thomison			
		Ausie Clement			
		Weaver Burton			
		Jim Clark			
		Hung Le			

REACTOR ENGINEER (CONTACT 1)

INITIAL	TIME CONTACTED		PAX	DIM	HOME
		Earl Nave			
		Mike Wingo			
		Bill Williamson			

SYSTEMS & TEST ENGINEER (CONTACT 1)

INITIAL	TIME CONTACTED		PAX	DIM	HOME
		Dwight Mims			
		R. McPherson			
		Paul Romine			

MECHANICAL ENGINEER (CONTACT 1)

INITIAL	TIME CONTACTED		PAX	DIM	HOME
		Tink Haney			
		Charlie Wages			
		Jim Walker			

Revision 0

Revision Log Sheet

Revision Date: MAR 24 1983

This log sheet must be retained as the last page of the Division of Nuclear Power Emergency Center Implementing Procedures Document.

Inserted by: ______________ Date Inserted: __________

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REP-IPD

DNPEC - IP-6

DIVISION OF NUCLEAR POWER EMERGENCY CENTER
PROCEDURE FOR
ALERT, SITE EMERGENCY, AND GENERAL EMERGENCY

Prepared By: W. E. Webb, Jr.

Approved By: [signature]

Date: 9/25/81

Rev. No.	Date	Revised Pages	Rev. No.	Date	Revised Pages
0	9/25/81	All	3	MAR 24 1983	All
1	3/10/82	2, 3, 4			
2	OCT 26 1982	All			

The last page of this procedure is Number 11.

DIVISION OF NUCLEAR POWER EMERGENCY CENTER

PROCEDURE FOR

ALERT, SITE AREA EMERGENCY, AND GENERAL EMERGENCY

1.0 PURPOSE

This procedure is designed to direct the Division of Nuclear Power Emergency Center (DNPEC) Director and staff to ensure a consistent, accurate, and timely response to the events of an accident. This procedure further serves to identify the necessary information which the DNPEC must provide the Central Emergency Control Center (CECC) Director to ensure that prompt, accurate, public protective action recommendations can be made by the CECC to appropriate State authorities.

2.0 SCOPE

This procedure covers the actions of the DNPEC Director and staff during an Alert, Site Area Emergency, or General Emergency.

3.0 REFERENCES

Radiological Emergency Plan.

4.0 ABBREVIATIONS AND DEFINITIONS

None.

5.0 RESPONSIBILITIES

5.1 In the event of a radiological emergency classified as an Alert, Site Area Emergency, or General Emergency, the Operations Duty Specialist (ODS) calls the Division of Nuclear Power (NUC PR) Emergency Duty Officer (EDO) to act as DNPEC and CECC Director until he is relieved by the permanent CECC Director and DNPEC Director. The ODS is also responsible for contacting the DNPEC staff and having them report to the DNPEC. After contacting the EDO, the ODS contacts the DNPEC Director and requests that he report to the DNPEC. Upon his arrival at the DNPEC, the DNPEC Director relieves the EDO of his duties as acting DNPEC Director. The DNPEC Director is responsibile for committing the support efforts of NUC PR to the affected plant. If NUC PR cannot fulfill the needs of the affected plant, the DNPEC Director has the authority to seek help from other divisions within TVA. The DNPEC Director is responsible for ensuring that the CECC is provided with the current summaries of information needed for overall accident assessment.

*Revision

5.2 DNPEC Staff

The DNPEC staff is responsible for assisting the DNPEC Director in carrying out the DNPEC responsibilities in providing NCO technical support to the affected plant and to the CECC. An assignment of positions and duties of this staff is as follows:

5.2.1 DNPEC Director

1. Provide support services to the plant by utilizing all of the necessary manpower and equipment under the control of the division.

2. Ensure that employees who may be required to go to the affected plant are fully briefed prior to leaving, and know to whom they are to report.

3. Keep the Site Emergency Organization informed of personnel ordered to the site and expected time of arrival.

4. Maintain current status reports and summaries of key events from the plant and provide this information to the CECC Director and other support organizations as needed.

*5. Responsible to the CECC Director to ensure that he is kept periodically briefed (at a minimum, hourly) and provided the necessary information pertaining to plant status for the overall assessment of the accident (see attachment 1).

6. Requests assistance from other divisions, local agencies, government installations, or vendors, as needed.

7. Maintains contact with the Site Emergency Director and ensures that necessary support is provided.

8. Assigns an individual to maintain a sequence of key events. This sequence of events is to be updated hourly.

5.2.2 ODS

1. Provides for initial notification of all offsite emergency organizations upon declaration of an emergency classification.

*Revision

2. Notifies all DNPEC and CECC staff members when required to report to the CECC.

3. Performs notifications to other organizations or personnel as requested by the DNPEC or CECC Directors.

5.2.3 Plant Communicator

1. Maintains continuous communications with the site Technical Support Center (TSC).

2. Responsible for ensuring the electronic blackboard is operational and kept current with the sequence of events relevant to the incident.

3. Responsible for receiving plant parameter lists, via the telecopy machine, and ensuring the information is promptly distributed to the DNPEC Director and staff.

4. Assists the DNPEC Director in other communication needs with the site as necessary.

5.2.4 KEC Communicator

1. Maintains continuous communications with the KEC and ensures current information is provided to them.

2. Responsible for coordinating all requests and responses made by the DNPEC to and from the KEC.

***3. Shall provide all KEC technical recommendations to the Reactor Engineering Branch assessment team for evaluation and consideration.

5.2.5 Reactor Engineering Branch Representative

1. Directs Reactor Engineering Branch activities in support of the plant.

2. Provides the DNPEC Director the required plant accident assessment information. This assessment shall address, as a minimum, the conditions shown in attachment 1 and shall be keyed to specific plant parameters. The DNPEC Director will use the assessment to brief the CECC (hourly), request assistance as needed from the KEC and other organizations, and to advise and confer with the Site Emergency Director.

***Addendum

3. Supports the DNPEC Director as required in coordinating emergency support actions with the site.

4. Provides logistics support as required to the DNPEC.

5.2.6 Electrical and Instrument and Controls (EI&C) Branch Representative

1. Directs EI&C Branch activities in support of the plant.

2. Supports the DNPEC Director as required in coordinating emergency support actions with the site.

3. Provides logistics support as required to the DNPEC.

4. Provides, as necessary, a representative to assist the Reactor Engineering Branch with the ongoing accident assessment.

5.2.7 Mechanical Branch Representative

1. Directs the Mechanical Branch activities in support of the plant.

2. Supports the DNPEC Director as required in coordinating emergency support actions with the site.

3. Provides logistics support as required to the DNPEC.

4. Provides, as necessary, a representative to assist the Reactor Engineering Branch with the ongoing accident assessment.

5.2.8 Field Services Branch Representative

1. Directs the Field Services Branch activities in support of the plant.

2. Supports the DNPEC Director as required in coordinating emergency support actions with the site.

3. Provides logistics support as required to the DNPEC.

4. Provides, as necessary, a representative to assist the Reactor Engineering Branch with the ongoing accident assessment.

*Revision

5.2.9 Emergency Preparedness and Protection (EP&P) Branch Representative

1. Directs the EP&P Branch activities in support of the plant.

2. Supports the DNPEC Director as required in coordinating emergency support actions with the site.

3. Assists, as necessary, the TSC Health Physics and Rad-Chem Supervisors in coordinating the performance of the predictive release rate model with the KEC and MSEC.

4. Responsible for providing current offsite information, as provided by the CECC, to the TSC and for notifying the DNPEC Director when such information is updated.

5. Provides logistics support as required to the DNPEC.

6. Provides, as necessary, a representative to assist the Reactor Engineering Branch with the ongoing accident assessment.

5.2.10 Management Services Staff Representative

1. Provides clerical staff for the DNPEC and CECC.

2. Makes appropriate arrangements for travel and lodging for emergency personnel as necessary.

3. Provides food for emergency personnel.

4. Ensures that the Document Control Center is manned and operational and coordinates the retrieval of manuals, prints, etc.

5. Provides logistics support as required to the DNPEC.

5.2.11 Nuclear Regulatory Commission (NRC)

The NRC role in the DNPEC is to observe and advise, as appropriate, with licensee decisions and actions.

5.2.12 State Representative

The State representative's role in the DNPEC is to observe events taking place and licensee actions and advise the State agencies appropriately throughout the emergency.

*Revision

5.2.13 Clerical Staff

1. Operates CRT terminals.

2. Maintains log of events.

3. Answers telephones.

4. Operates the Dimension telephone console.

5. Maintains DNPEC Organization Board.

6. Operates telecopy machine.

7. Performs other duties as required by the DNPEC staff.

5.3 Supporting Divisions

If necessary, the DNPEC Director may obtain assistance from other divisions within TVA.

5.4 Technical Support

The technical support group is composed of skilled professionals trained to provide plant support. The group consists of TVA central office employees in NUC PR. The technical support members provide expertise in reactor systems and core engineering, electrical engineering, mechanical engineering, chemical engineering, chemistry, shielding, transient analysis, fire protection, electrical distribution (inplant), security, metallurgy, radwaste, and instrumentation. The DNPEC Director delegates to the lead DNPEC staff representative the job of contacting these designated individuals by phone as needed. If necessary, technical support personnel may be sent to the plant. Transportation may be provided by helicopter or fixed-wing aircraft.

6.0 PROCEDURE REQUIREMENTS

NOTE: The EDO will follow this procedure until relieved by the permanent DNPEC Director. The permanent DNPEC Director debriefs the EDO and continues with the procedure where the EDO left off.

6.1 Notifications

6.1.1 Verify that the ODS has contacted any requested technical support personnel. If technical support personnel are to be sent to the site by aircraft, verify that the ODS has made appropriate arrangements.

*Revision

6.1.2 Notify the Site Emergency Director that the DNPEC is activated and obtain a current status report.

6.1.3 Review the emergency condition with the CECC Director.

6.1.4 Establish communications with the KEC. Designate an individual from the DNPEC staff to act as primary communications contact with the KEC until a Knoxville employee can staff the position.

Alternate phone numbers are listed in the TVA Radiological Emergency Notification Directory.

6.2 Accident Assessment

6.2.1 Verify that the electronic blackboard is manned and operational and is being kept current with the synoptics of the emergency (Plant Communicator's responsibility).

*6.2.2 Verify that the key plant parameter lists are being transmitted to the DNPEC via telecopy, and are promptly distributed to the DNPEC staff, plant assessment team, and CECC.

6.2.3 Verify that an individual has been designated to construct and maintain a current chronological key sequence of events. Ensure this information is transmitted to the CECC and KEC.

6.2.4 Verify that the Reactor Engineering Branch assessment team continually performs a plant accident assessment and that they make prompt reports to the DNPEC Director. Evaluations must be keyed to specific plant parameters.

***6.2.5 Formal recommendations from the assessment team shall be approved by the DNPEC Director and provided by him to the Site Emergency Director.

*6.2.6 The DNPEC Director shall ensure the accident assessment information is provided to the CECC on a frequent basis, (at a minimum, hourly) and he shall periodically brief the CECC Director on plant status and advise him in overall accident assessment (see attachment 2).

*6.2.7 Verify that the Emergency Preparedness and Protection Branch representative is providing current offsite assessment information to the TSC.

*Revision
***Addendum

***6.2.8 Potential Release Evaluation

If after consultation with the CECC, the CECC Director requests that a predictive release evaluation be performed based on the potential for significant changes in plant conditions, the DNPEC shall provide the KEC the appropriate assumptions to be made pertaining to plant status for performing the necessary calculations. The areas to be considered are as follows:

a. Increased fuel failure (changes in primary coolant activity levels).

b. Anticipated changes in primary coolant leakage rates or break sizes.

c. Anticipated changes in containment leakage rates (i.e., changes in containment pressure and/or changes in size of containment ruptures or holes).

6.3 <u>General Operation</u>

6.3.1 During the course of an emergency, should the accident upgrade, downgrade, or terminate, the DNPEC Director shall notify the KEC and CECC Directors immediately.

6.3.2 The DNPEC Director shall conduct periodic briefings for the entire DNPEC staff to update the emergency situation.

6.3.3 If available personnel and equipment of NUC PR are not enough to cope with the emergency, contact the designated representative of other TVA divisions, as necessary, to supply adequate resources to recover from the accident. Log the organizations called for assistance. A description of services available and emergency contacts are available in the TVA Radiological Emergency Notification Directory (REND).

6.3.4 For a Site Area or General Emergency, the site should be reminded that additional technical personnel are available from the DNPEC to upgrade the technical support capability at the TSC. The DNPEC Director should discuss the need for this upgraded capability with the Site Emergency Director. Based upon this discussion, selected technical support personnel may be dispatched by ground or air transportation.

*Revision
***Addendum

6.3.5 Relief of Duties--Should the accident be expected to last for an extended period, the DNPEC Director originates a schedule for relief. The duties of DNPEC Director should only be passed on to individuals identified as alternates for the DNPEC Director's position. However, for short periods of time, the DNPEC Director may delegate the authority of DNPEC Director to a member of the DNPEC staff until an alternate DNPEC Director can arrive. The DNPEC Director also directs his staff to prepare a schedule for their relief to ensure the necessary staff of the DNPEC is available for the duration of the emergency. The DNPEC Director gives the Management Services representative a copy of the schedule so that he may notify the individuals of the time they are to report.

6.3.6 The DNPEC Director and staff will support the CECC Director as required for carrying out recovery efforts from the accident.

6.3.7 Upon termination of the emergency, the DNPEC Director and staff shall make themselves available for review of the accident.

*Revision

ATTACHMENT 1

ACCIDENT ASSESSMENT CECC STATUS BOARD

I. HEAT REMOVAL CAPABILITY:

II. FUEL INTEGRITY:

III. RADIOACTIVITY IN CONTAINMENT:

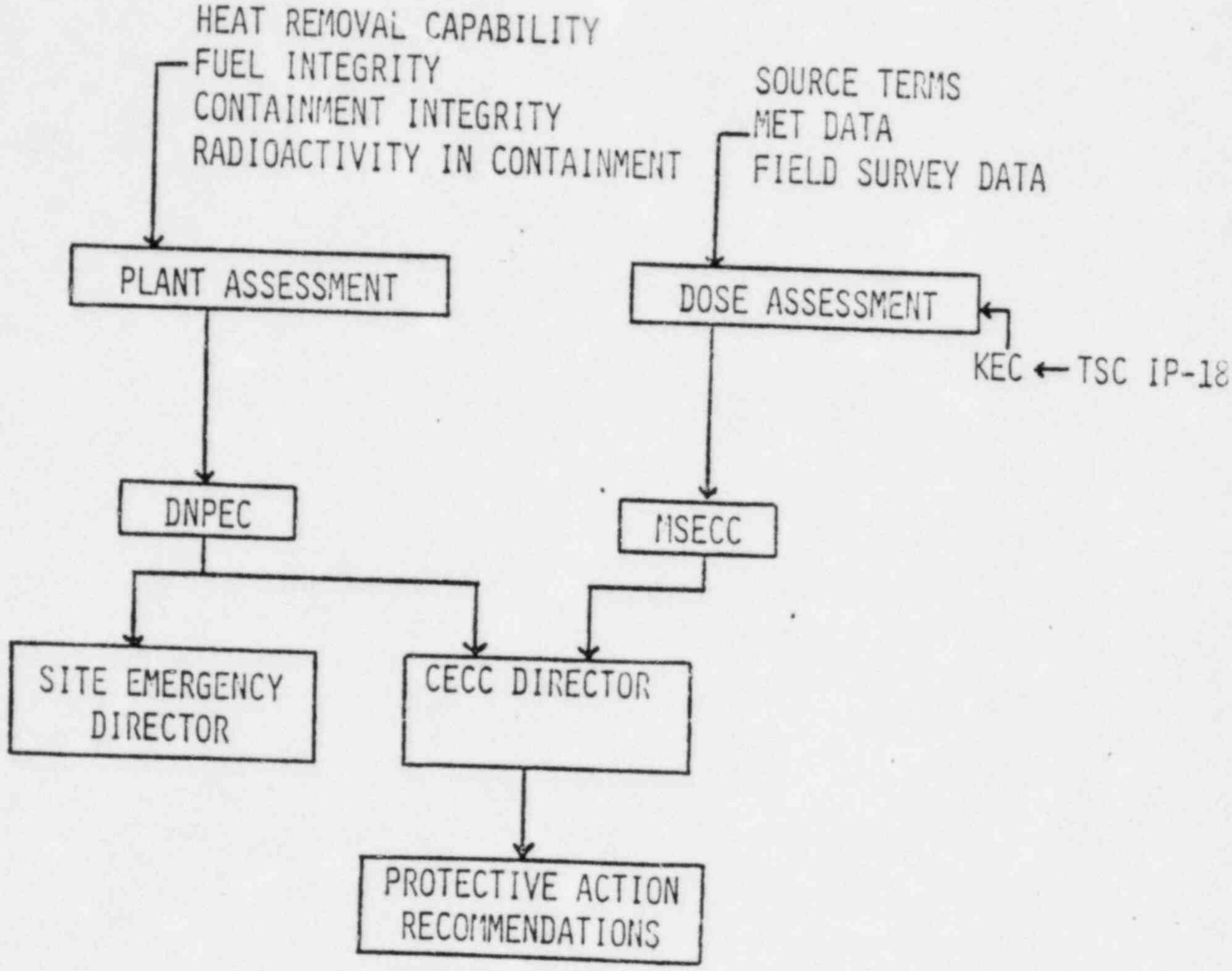
IV. CONTAINMENT INTEGRITY:

V. OVERALL ASSESSMENT

*10

*Revision

INPUT FOR CECC ACCIDENT ASSESSMENT



*Revision

REVISION LOG SHEET

Revision Date: FEB 23 1983 (issued 4/18/83)

This log sheet must be retained as the last page of the Knoxville Emergency Center Implementing Procedures Document.

Inserted by: ____________________ Date Inserted: ____________________

Pages to be Removed			New Pages to be Inserted		
Part	Page Number	Revision	Part	Page Number	Revision
			IP-6	Cover Page	Rev. 0
				1 of 3	Rev. 0
				2 of 3	Rev. 0
				3 of 3	Rev. 0
			Appendix A	1 of 79	Rev. 0
				thru	
				79 of 79	Rev. 0
			Appendix B	1 of 53	Rev. 0
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				53 of 53	Rev. 0
			Appendix C	1 of 50	Rev. 0
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				50 of 50	Rev. 0

EC-IPD

KEC IP-6

KEC OPERATING PROCEDURES

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Reviewed by: [signature]

Approved by: John A. [signature]

Date: FEB 23 1983

Revision Log

Rev No.	Date	Revised Pages

The last page of this procedure is No. 185

1.0 Purpose

This procedure is a consolidation of the methodology and formulae used by Knoxville Emergency Center (KEC) support personnel to evaluate and predict plant radionuclide releases in gaseous effluents at all TVA nuclear facilities in the event of a loss-of-coolant accident (LOCA). Each plant has its own appendix with plant-specific requirements included.

2.0 Scope

Each plant-specific appendix is to be used to predict noble gas and iodine effluent releases from its respective plant. The number of plant specific appendixes will be increased as new plants are added to the scope of the radiological emergency plan.

3.0 Responsibilities

Upon indication that an emergency situation exists at any of TVA's operating nuclear plants, that plant's Technical Support Center (TSC) personnel shall be responsible for data collection and release rate calculations until the KEC is staffed. Once staffed, the KEC will assume lead responsibility for the calculation of release potential. Data for the KEC evaluation must be updated on a 30-minute interval by designated site personnel. After the KEC results and assumptions have been coordinated with the applicable TSC, the KEC shall transmit the results to the Muscle Shoals Emergency Center for subsequent off-site dose calculations.

DE00:KECREP

Predictive Models

BACKGROUND

A loss of coolant accident (LOCA) normally results in radioactive material being released to the containment atmosphere.

With time, some of this material is released from containment and enters the environment. These releases will be monitored and, to the extent possible, must be predicted for proper management of the accident. A procedure in the Implementing Procedures Document for the KECC enables the user to:

1. Select and calculate existing plant parameters needed to calculate releases of radioactivity to the atmosphere, i.e.:

   a. Which radioactive isotope spectrum should be used?

   b. How severe is the fuel failure?

   c. How fast is radioactive coolant leaking into the containment structure?

   d. How fast is the radioactive material leaking from the containment structure?

2. Predict probable future releases of radioactive gases and vapors to the outside atmosphere, assuming plant conditions do not get worse.

3. Predict probable future releases of radioactive gases and vapors to the outside atmosphere, based on a plausible estimate of worsening plant conditions.

   The data and prediction criteria in this procedure are based on design information, anticipated operating procedures, and the fact that the amount and rate of radioactivity released to the environment depends entirely on the activity level inside containment, the leakage from containment, and the operation of equipment designed to reduce the effects of the leakage.

ASSUMPTIONS

1. ACTIVITY LEVEL

   Whatever the initiating event, it is assumed that primary coolant leaks into containment.

   This coolant is contaminated with several radioisotopes and can be classified according to its isotopic distribution. For each plant, at least two different distributions are used in the predictive model.

2. CONTAINMENT LEAKAGE

Radioactive gas leakage from containment can be expected to vary with the pressure drop across the containment walls. Containment leak tests are not considered adequate to establish this relationship, nor is it likely that a simple relationship would be found even if adequate data were available. For purposes of this plan, it is assumed that containment leakage is described by a simple orifice equation; the curves shown for containment leak rate versus pressure are based on this equation. These curves are included only for making predictions, since the calculated releases assume a leak rate from containment.

3. SAFETY SYSTEMS

The final step in determining the radioactivity releases following an accident considers the operation of engineered safety features. The calculations in this procedure take account of the possible locations where leakage could occur, as well as airflow patterns established after these engineered safety features start operation.

METHODOLOGY

The TVA method for predicting releases of radioactivity to the environment is based on evaluation of plant conditions (actual, suspected, or postulated) and comparison with a range of calculated results for certain assumed conditions. (The assumed conditions range from a best estimate or expected value to a "worst case" value.) The parameters examined are primary coolant activity, radioactivity discharge to containment, and leakage from containment. The effects of engineered safety features on releases are taken into account. The calculated results are reduced to graphical form which facilitates interpolation and projection.

1.0 PLANT PARAMETERS REQUIRED FOR RELEASE RATE CALCULATION

1.1 Determine which isotope spectrum to use based on the latest primary coolant measurement.

1.1.1 On worksheet 1-A record the latest measured primary coolant activity (per SI 4.6.B or postaccident measurement) and calculate the activity ratios.

1.1.2 On Table 1.1, circle the entries corresponding to the ratios calculated in 1.1.1. Circle the isotope spectrum on the line with the most circled entries.

1.1.3 Check the spectrum circled in step 1.1.2. This isotope spectrum will be used in all steps of 1.0 and 2.0 of this procedure.

IMPORTANT: A POWER CHANGE OF MORE THAN 15 PERCENT WITHIN 2-8 HOURS BEFORE THE PRIMARY COOLANT SAMPLE WAS TAKEN INVALIDATES THE RATIOS INVOLVING IODINE (I). IN SUCH A CASE, RELY ON THE C/S RATIO.

1.2 Determine normalization factor

The information labeled "NUREG-0016" is based on that amount of failed fuel which produces an off-gas release rate of 50,000 Ci/s at 30 minute decay. The "Modified TID-14844" information is based on 1.0×10^{-6} percent of the total radioactivity inventory of an equilibrium reactor core.

1.2.1 On worksheet 1-A, circle the column of calculated specific activities for the isotopes I-131, CS-137, and Sr-92 given in Table 1.2 for the isotope spectrum chosen in step 1.1.3.

1.2.2 Calculate and record the ratio of the measured specific activity (from step 1.1.1) to the circled calculated values in Table 1.2.

1.2.3 Select a normalization factor corresponding most closely to two of the three ratios. Record this on worksheet 1-A, step 1.2.3.

1.3 Determine primary system break size

Use worksheet 1-B. Record the readings of radiation monitors RE-90-272A and RE-90-273A (drywell radiation) and RE-90-272B and RE-90-273B (torus radiation). Also record date and time at which readings were taken. It should be noted that all radiation monitor readings are assumed to be background subtracted before being used in this predictive model. Readings taken immediately prior to the accident should be used as the background values if available. This applies to effluent as well as area monitors.

1.3.1 Calculate the elapsed time from reactor shutdown to the time at which the monitor readings were taken.

1.3.2 Record the normalization factor, F, determined in 1.2.3.

1.3.3 Normalize the measured dose rates by dividing by the normalization factor. The terminology used is: MDR = Measured Dose Rate; MDRN = Normalized Measured Dose Rate.

1.3.4 Determine whether the break has occurred in a recirculation line or in a steam line. (Initial system depressurization is faster for steam line breaks; containment temperature will exceed saturation temperature after approximately 5 minutes for steam line breaks.) Check the appropriate break type.

Proceed with step 1.3.5.

- OPTIONAL -

If the TID-14844 source spectrum was chosen in 1.1.3, the following method may be used to determine the break type (This should be used only if the break type cannot be determined any other way.):

1.3.4.1 Calculate the ratio of drywell radiation, D(mR/hr) to torus radiation, T(mR/hr). Use the monitor readings recorded in 1.3. Record the ratio below.

Ratio: D/T = ________ mR/hr ________ mR/hr = ________

RE-90-272A	RE-90-272B	(D/T)
or	or	
RE-90-273A	RE-90-273B	

1.3.4.2 If the drywell to torus radiation ratio (D/T) calculated above is greater than 1.0 but less than 10.0 (i.e., 1.0 D/T 10.0), then it can be assumed that the break has occurred in a steam line. For a D/T ratio of anything else (i.e., D/T 1.0 or D/T 10.0), a break in a recirculation line can be assumed. Proceed with step 1.3.5.

Check One: ____ 1.0 D/T 10.0 (Steam Line Break)
____ D/T 1.0 or D/T 10.0 (Recirculation Line Break)

1.3.5 The break size will be determined using the normalized value of the drywell radiation determined in 1.3.3. If these monitors cannot be used, skip to 1.3.6.

1.3.5.1 Record the normalized drywell radiation (mR/hr), determined in 1.3.3. Select the greater of the two MDRN (drywell) recorded in 1.3.3. Note that the two drywell monitors (RE-90-272A and RE-90-273A) should have similar readings. Large deviations may imply that one of the monitors is not operating properly, in which case the reading from the operating monitor should be used. In any case, the graphs which are labeled as monitor RE-90-272A will be used.

1.3.5.2 Choose the appropriate figure to be used for visual interpolation using the source spectrum from 1.1.3 and the break type from 1.3.4:

Recirculation Break/NUREG-0016 Spectrum: Figure A.1-1
Steam Line Break/NUREG-0016 Spectrum: Figure A.1-2
Recirculation Break/TID-14844 Spectrum: Figure A.1-3
Steam Line Break/TID-14844 Spectrum: Figure A.1-4

1.3.5.3 Using the elapsed time t_e from 1.3.1, and the normalized drywell radiation (mR/hr), estimate a primary system break size by visually interpolating between the curves in the figure chosen in 1.3.5.2. Record the value of the calculated monitor reading and the associated break size from the curves above and below the normalized measured value.

1.3.5.4 Record the estimated primary system break size on worksheet 1-B. Proceed to step 1.4.

1.3.6 The primary system break size will be determined using the normalized value of the torus radiation monitor determined in 1.3.3 only if monitors RE-90-272A and RE-90-273A (drywell radiation) cannot be used.

1.3.6.1 Record the normalized torus radiation (mR/hr), determined in 1.3.3. Select the greater of the two MDRN (Torus) recorded in 1.3.3. Note that the two torus monitors (RE-90-272B and RE-90-273B) should give similar readings. Large deviations may imply that one of the monitors is not operating properly, in which case the reading from the operating monitor should be used. In any case, the graphs which are labeled as monitor RE-90-272B will be used.

1.3.6.2 Choose the appropriate figure to be used for visual interpolation using the source spectrum from 1.1.3 and the break type from 1.3.4.

Recirculation Break/NUREG-0016 Spectrum: Figure A.2-1
Steam Line Break/NUREG-0016 Spectrum: Figure A.2-2
Recirculation Break/TID-14844 Spectrum: Figure A.2-3
Steam Line Break/TID-14844 Spectrum: Figure A.2-4

1.3.6.3 Using the elapsed time t_e from 1.3.1, and the normalized torus radiation (mR/hr), estimate a primary system break size by visually interpolating between the curves in the figure chosen in 1.3.6.2. Record the value of the calculated monitor reading from the curves above and below the normalized measured value.

1.3.6.4 Record the estimated primary system break size on worksheet 1-B. Proceed to step 1.4.

1.4 Determine Containment Leak Rate

Use worksheet 1-C.

Record the readings of radiation monitors RE-90-147 and RE-90-148 (stack radiation). Record the time and date at which the readings were taken. Also, record the containment pressure.

Note: If the readings of the stack radiation monitors are not significantly greater than the normal background count rate, the containment leak rate cannot be determined analytically. For this situation, the containment leak rate will be assumed to be 1100 scfh (2 percent per day containment free volume at 49.6 psig). If this assumption is made, skip to 1.4.6.

1.4.1 Calculate the indicated radioactivity release rate by multiplying the count rate (cps) of the stack gas radiation monitors by 24.7 μCi/sec/cps.

1.4.2 Calculate the elapsed time, t_e, from reactor shutdown to the time at which the readings were taken.

1.4.3 Record the normalization factor F, determined in 1.2.3.

1.4.4 Normalize the larger of the two release rates determined in 1.4.1 by dividing by the normalization factor. The terminology used is: SRRN = Normalized stack release rate.

1.4.5 Select, from Figure A.4, the figure to be used for visual interpolation based on the isotopic spectrum and primary system breaksize and type determined in 1.1 and 1.3. Record the figure number selected in 1.4.5, worksheet 1-C.

1.4.6 Using the elapsed time and the normalized release rate, estimate a gaseous containment leak rate by visually interpolating between the curves of the appropriate Figure A.4. Record the value of the calculated stack release rate and the associated containment leak rate from the curves above and below the normalized measured value from 1.4.4.

1.4.7 Record the estimated containment leak rate in 1.4.7 on worksheet 1-c.

1.4.8 Using the containment pressure from step 1.4 and the estimated containment leak rate, use Figure A.3 to obtain an equivalent hole size. Record the hole size diameter (inches) in 1.4.8 on worksheet 1-c.

2.0 PREDICTED RELEASE RATE--NO CHANGE IN PLANT CONDITIONS

2.1 Calculate the stack noble gas release rate

Use worksheet 2-A.

2.1.1 Use the same sheet of Figure A.4 selected in Step 1.4.5.

2.1.2 Use the containment leak rate LRD recorded in Step 1.4.7.

2.1.3 Select the time for which the stack release rate is desired.

2.1.4 Obtain the stack noble gas release rate by interpolation between the curves on Figure A.4:

2.1.4.1 Determine interpolation fraction = $\frac{LRD - LR(L)}{LR(U) - LR(L)}$

where: LRD = containment leak rate from Step 1.4.7.
LR = containment leak rate for which curve is calculated.
(U),(L) = value of LR immediately (above) (below) LRD.

2.1.5 Determine the time elapsed from reactor shutdown to T.

2.1.6 Determine the difference between stack release rates corresponding to LR(U) and LR(L) at the desired time, multiply by the interpolation fraction, and add to the vent release rate corresponding to LR(L). Then multiply this quantity by the normalization factor F.

2.2 Calculate the stack iodine release rate

Use worksheet 2-B.

2.2.1 Follow procedure 2.1.6. Substitute Figure A.5 for Figure A.4.

NOTE: It is not necessary to recalculate the interpolation fraction (Step 2.1.4.1) since it will be unchanged.

2.3 Calculate the turbine building noble gas release rate

Use worksheet 2-B.

2.3.1 Follow procedure 2.1.6. Substitute Figure A.6 for Figure A.4.

NOTE: It is not necessary to recalculate the interpolation fraction (Step 2.1.4.1) since it will be unchanged.

2.4 Calculate the turbine building iodine release rate

Use worksheet 2-C.

2.4.1 Follow procedure 2.1.6. Substitute Figure A.7 for Figure A.4.

NOTE: It is not necessary to recalculate the interpolation fraction (Step 2.1.4.1) since it will be unchanged.

3.0 PREDICTED RELEASE RATE--CHANGING PLANT CONDITIONS

3.1 For a different postulated primary coolant activity:

3.1.1 Consider which isotope spectrum ("NUREG-0016" or "Modified TID-14844") was chosen in Step 1.1.

3.1.2 Consider the magnitude of the normalization factor determined in Step 1.2.

3.1.3 Decide on a new primary coolant activity by increasing the normalization factor or changing the istope spectrum based on plausible developments in the condition of the plant. In order of increasing severity, isotope spectra rank as follows:

(1) NUREG-0016

(2) Modified TID-14844

If the isotope spectrum is "NUREG-0016," do not increase the normalization factor by more than 100; instead, change the spectrum to "Modified TID-14844."

3.2 <u>For a different postulated containment leak rate</u>

3.2.1 For an anticipated change in containment pressure with no additional containment degradation, use Figure A.3 with the equivalent hole size determined in Step 1.4.8.

3.2.2 For anticipated containment degradation, use Figure A.3 with an appropriately chosen hole size.

3.2.3 Determine the new containment leak rate from Figure A.3.

3.3 <u>For an anticipated degradation of the primary system</u>

Change the primary system break size.

3.4 <u>For the postulated changed plant parameters, recalculate future vent releases using the methods given in part 2.0</u>

NOTE: Part 2.0 is based on unchanged plant parameters. If the postulated conditions are more severe than those determined to exist in part 1.0, the part 2.0 methods with the changed parameters will result in release rates which are too high; conversely, in the unlikely event that the postulated conditions are less severe, the estimated release rates will be too low.

GASEOUS RELEASES WORKSHEET

1-A

1.1 Determination of Isotopic Spectrum

1.1.1 Im = I-131 Specific Activity ______________ μCi/g

Cm = Cs-137 Specific Activity ______________ μCi/g

Sm = Sr-131 Specific Activity ______________ μCi/g

Im/Cm = __________ Im/Sm = __________ Cm/Sm = __________

1.1.2

TABLE 1.1
SELECTION OF ISOTOPE SPECTRUM

I/C	I/S	C/S	Isotope Spectrum
I/C 10.0	I/S 20.0	C/S 1.0	NUREG-0016
I/C 10.0	I/S 20.0	C/S 1.0	Modified TID-14844

1.1.3 CHOSEN DISTRIBUTION IS (CHECK ONE)

______ "NUREG-0016"

______ "Modified TID-14844"

1.2 Determination of Normalization Factor

1.2.1

TABLE 1.2
SPECIFIC ACTIVITY IN PRIMARY COOLANT (μCi/g)

Isotope	NUREG-0016	Modified TID-14844
Ic	3.7×10^{-3}	3.2×10^{-3}
Cc	8.0×10^{-5}	3.6×10^{-3}
Sc	1.0×10^{-2}	5.0×10^{-5}

1.2.2 Subscript m Denotes Measured Specific Activity
Subscript c Denotes Calculated Specific Activity From Table 1.2

Im/Ic = __________ Cm/Cc = __________ Sm/Sc = __________

GASEOUS RELEASES WORKSHEET

1-A (CONTINUED)

1.2.3 Select A Normalization Factor Corresponding Most Closely To Two of the Three Ratios:

NORMALIZATION FACTOR F = ________

GASEOUS RELEASES WORKSHEET

1-B

1.3 Determine Primary System Break Size

Radiation Monitor Readings at ________________
(Time and Date)

RE-90-272A (Drywell): __________ mR/hr = MDR (Drywell)

RE-90-273A (Drywell): __________ mR/hr = MDR (Drywell)

RE-90-272B (Torus) : __________ mR/hr = MDR (Torus)

RE-90-273B (Torus) : __________ mR/hr = MDR (Torus)

1.3.1 Reactor Shutdown Date and Time ________________

Monitor Reading Date and Time ________________

Elapsed Time t_e = _____ hr ___ min ___ = ___ hr

1.3.2 Normalization Factor (from 1.2.3, worksheet 1-A)

1.3.3 Normalized Dose Rates

MDRN (Drywell) = MDR/F = RE-90-272A/F = __________ mR/hr

MDRN (Drywell) = MDR/F = RE-90-273A/F = __________ mR/hr

MDRN (Torus) = MDR/F = RE-90-272B/F = __________ mR/hr

MDRN (Torus) = MDR/F = RE-90-273B/F = __________ mR/hr

1.3.4 Break Type (Check One)

____________ Recirculation Line Break

____________ Steam Line Break

1.3.5 Determination of Break Size Using Drywell Monitor Values

1.3.5.1 MDRN (Drywell) From 1.3.3 = ______________ mR/hr
Where MDRN is the normalized dose rate at t_e

1.3.5.2 Calculated Monitor Reading from Fig. A.1 ____ (Record figure used)

GASEOUS RELEASES WORKSHEET

1-B (Continued)

1.3.5.3 Elapsed Time t_e from 1.3.1 = ______ hr

MDRN (drywell) from 1.3.3 = ____ mR/hr

DCDR(U) = ____ mR/hr; upper break size = ____ ft^2

DCDR(L) = ____ mR/hr; lower break size = ____ ft^2

Where: DCDR(U) = Calculated drywell monitor reading greater than MDRN (Drywell)
DCDR(L) = Calculated drywell monitor reading lower than MDRN (Drywell)

1.3.5.4 Primary system break size: _______ ft^2

1.3.6 Determination of Break Size Using Torus Monitor Values

1.3.6.1 MDRN (Torus) from 1.3.3 = ____________ mR/hr
Where MDRN is the normalized dose rate at t_e

1.3.6.2 Calculated monitor reading from Fig. A.2 ___ (Record figure used)

1.3.6.3 Elapsed Time t_e from 1.3.1 = _______ hr

MDRN (Torus) from 1.3.3 = _____ mR/hr

TCDR(U) = ____ mR/hr; upper break size = ____ ft^2

TCDR(L) = ____ mR/hr; lower break size = ____ ft^2

Where: TCDR(U) = Calculated torus monitor reading greater than MDRN(TORUS)

TCDR(L) = Calculated torus monitor reading lower than MDRN(TORUS)

1.3.6.4 Primary system break size: ________ ft^2

GASEOUS RELEASES WORKSHEET

1-C

1.4 Determination of Containment Leak Rate

Radiation Monitor Readings at ______ ______ (Time and Date)

RE-90-147: ________ cps

RE-90-148: ________ cps

Containment Pressure:

PI-64-67B: ________ lb/in^2

PI-64-50A: ________ lb/in^2

Time Since Reactor Shutdown = _____ hrs _____ min = _____ hrs

1.4.1 SRR(1) = ________ (RE-90-147) cps x 24.7 μCi/s/cps = ________ μCi/s

SRR(2) = ________ (RE-90-148) cps x 24.7 μCi/s/cps = ________ μCi/s

Where: SRR(1) = Stack release rate determined by RE-90-147
SRR(2) = Stack release rate determined by RE-90-148

1.4.2 Time Since Reactor Shutdown = _____ hrs _____ min = _____ hrs

1.4.3 Normalization Factor, F = ________

1.4.4 SRRN = ________ (SRR(1) or SRR(2)) μCi/s ÷ _____ (F) = _____ μCi/s

1.4.5 Figure Used for Visual Interpolation: Figure A.4 ________

1.4.6 CRR(U) = ____ μCi/s; corresponding leak rate ________ SCFH

CRR(L) = ____ μCi/s; corresponding leak rate ________ SCFH

Where: CRR(U) = Calculated stack release rate greater than SRRN
CRR(L) = Calculated stack release rate lower than SRRN

1.4.7 Estimated Containment Leak Rate

LRD = ________ SCFH

1.4.8 Equivalent Hole Size Diameter = ________ in.

GASEOUS RELEASES WORKSHEET

2-A

2.0 PREDICTED RELEASE RATE--NO CHANGE IN PLANT CONDITIONS

2.1 Calculate the Stack Noble Gas Release Rate

2.1.1 From Figure A.4 _____ (Record Figure Used)

LR(U) = ________ SCFH; LR(L) = ________ SCFH

Where: LR(U) = Value of Leak Rate Immediately Above LRD
LR(L) = Value of Leak Rate Immediately Below LRD

2.1.2 LRD = ________ SCFH

2.1.3 Time for which release rate is desired

T = __________

2.1.4 Interpolation Fraction

$$IF = \frac{LRD - LR(L)}{LR(U) - LR(L)}$$

$$IF = \frac{(\quad) - (\quad)}{(\quad) - (\quad)} = \_\_\_\_\_\_\_\_$$

2.1.5 Time Elapsed from Reactor Shutdown to T

$t = t_e =$ ______ hrs ____ min = ____ hrs

2.1.6 From Fig. A.4, At t, Determine CRR(U) Corresponding to LR(U)

CRR(U) = __________ μCi/s

From Fig. A.4, At t, Determine CRR(L) Corresponding to LR(L)

CRR(L) = __________ μCi/s

Normalization Factor (From 1.2.3): F = ______

Future Noble Gas Release Rate =

[IF x ((CRR(U) - CRR(L)) + CRR(L)] x F

[_____ x (_____ - _____) + _____] x _____

= _____ μCi/s

GASEOUS RELEASES WORKSHEET

2-B

2.2 Calculate the Stack Iodine Release Rate

2.2.1 Record Interpolation Fraction, IF, From 2.1.4

IF = ______

From Fig. A.5 ______ (Record Figure Used)

CRR(U) = ______ μCi/s

CRR(L) = ______ μCi/s

Normalization Factor (From 1.2.3): F = ______

Future Iodine Release Rate =

[IF x ((CRR(U) - CRR(L)) + CRR(L)] x F

[_____ x (_____ - _____) + _____] x _____

= ______ μCi/S

2.3 Calculate the Turbine Building Noble Gas Release Rate

2.3.1 Record Interpolation Fraction, IF, From 2.1.4

IF = ______

From Fig. A.6 ______ (Record Figure Used)

CRR(U) = ______ μCi/s

CRR(L) = ______ μCi/s

Normalization Factor (From 1.2.3): F = ______

Future Noble Gas Release Rate =

[IF x ((CRR(U) - CRR(L)) + CRR(L)] x F

= [_____ x (_____ - _____) + _____] x _____

= ______ μCi/s

GASEOUS RELEASES WORKSHEET

2-C

2.4 Calculate the Turbine Building Iodine Release Rate

2.4.1 Record Interpolation Fraction, IF, From 2.1.4

IF = _______

From Fig. A.7. _______ (Record Figure Used)

CRR(U) = _______ μCi/s

CRR(L) = _______ μCi/s

Normalization Factor (From 1.2.3): F = _______

Future Iodine Release Rate =

[IF x (CRR(U) - CRR(L)) + CRR(L)] x F

= [_____ x(_____ - _____) + _____] x _____

= _______ μCi/s

FIGURES

FIGURES A.1

FIGURES A.2

FIGURE A.3

A.3 Leak Rate versus Pressure for Differing Hole Size Diameters

FIGURES A.4

FIGURES A.5

FIGURES A.6

*TB = Turbine Building

FIGURES A.7

*TB = Turbine Building

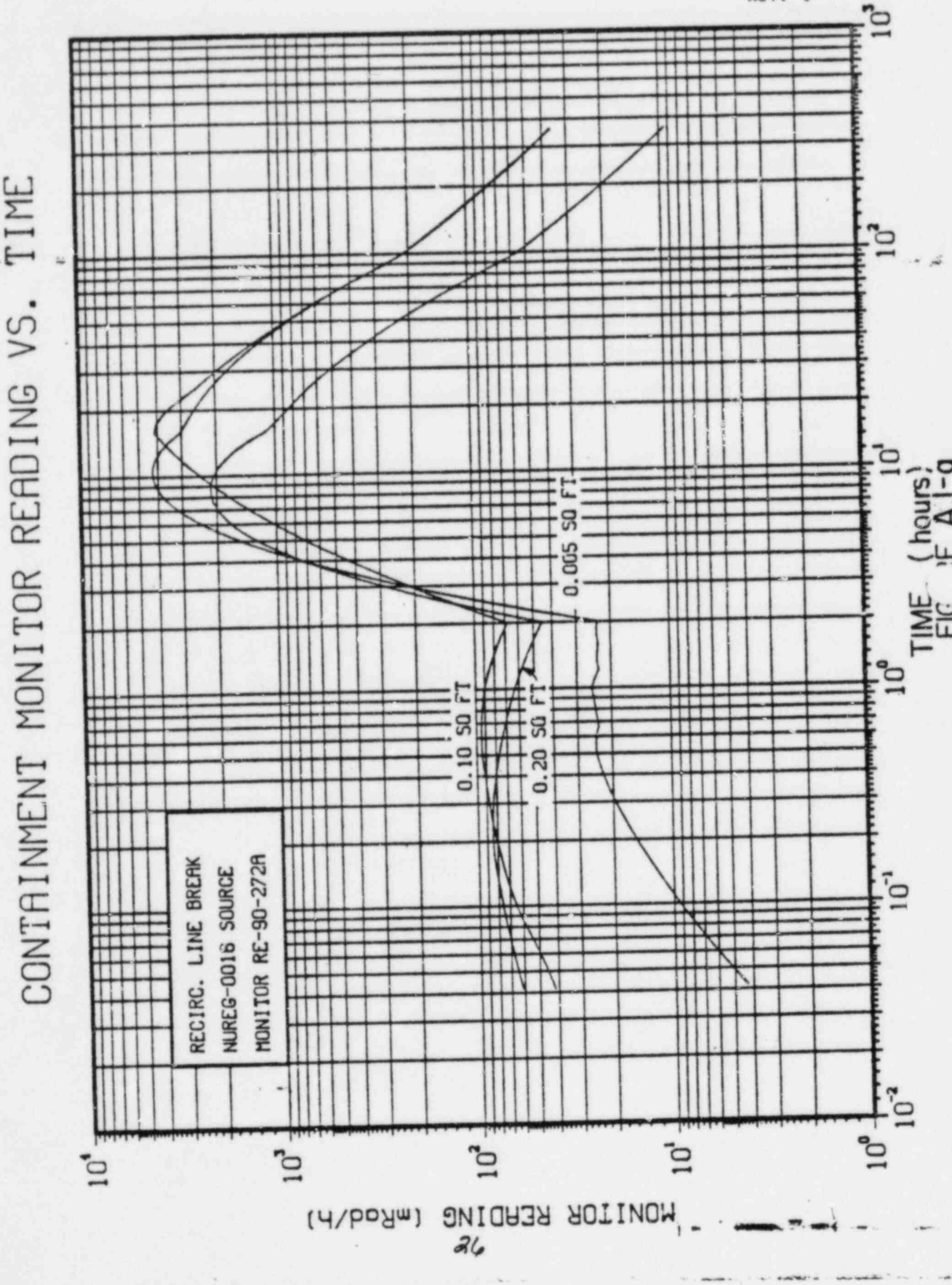
CONTAINMENT MONITOR READING VS. TIME
RECIRC. LINE BREAK
NUREG-0016 SOURCE
MONITOR RE-90-272A
0.005 SQ FT
0.10 SQ FT
0.20 SQ FT
MONITOR READING (mRad/h)
TIME (hours)
10^{4}
10^{3}
10^{2}
10^{1}
10^{0}
10^{-2}
10^{-1}
10^{0}
10^{1}
10^{2}
10^{3}

FIGURE A.1-a

CONTAINMENT MONITOR READING VS. TIME

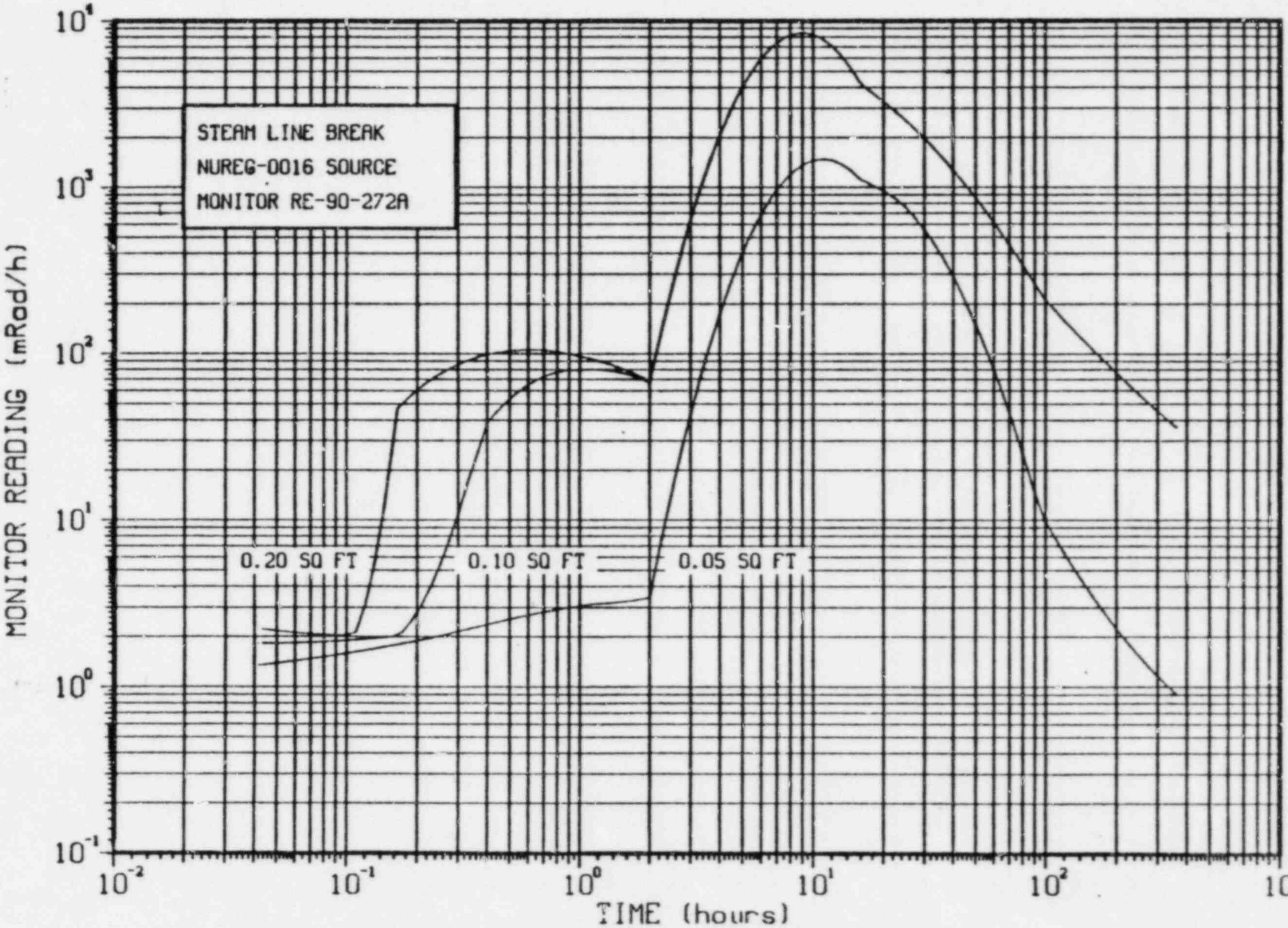

FIGURE A.1-b

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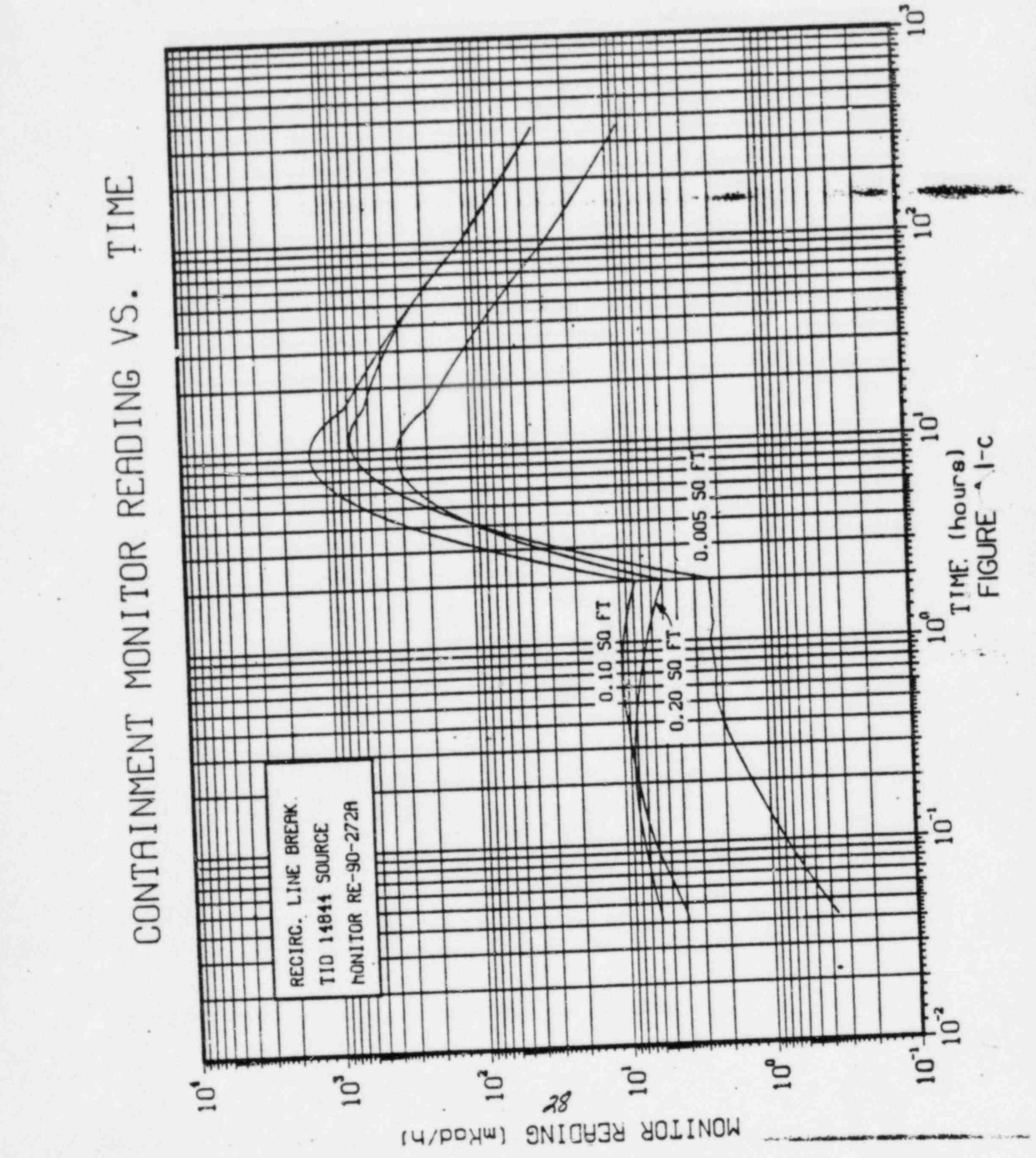

CONTAINMENT MONITOR READING VS. TIME
RECIRC. LINE BREAK
TID 14844 SOURCE
MONITOR RE-90-272A
0.10 SQ FT
0.20 SQ FT
0.005 SQ FT
MONITOR READING (mRad/h)
TIME (hours)
FIGURE A.1-c

CONTAINMENT MONITOR READING VS. TIME

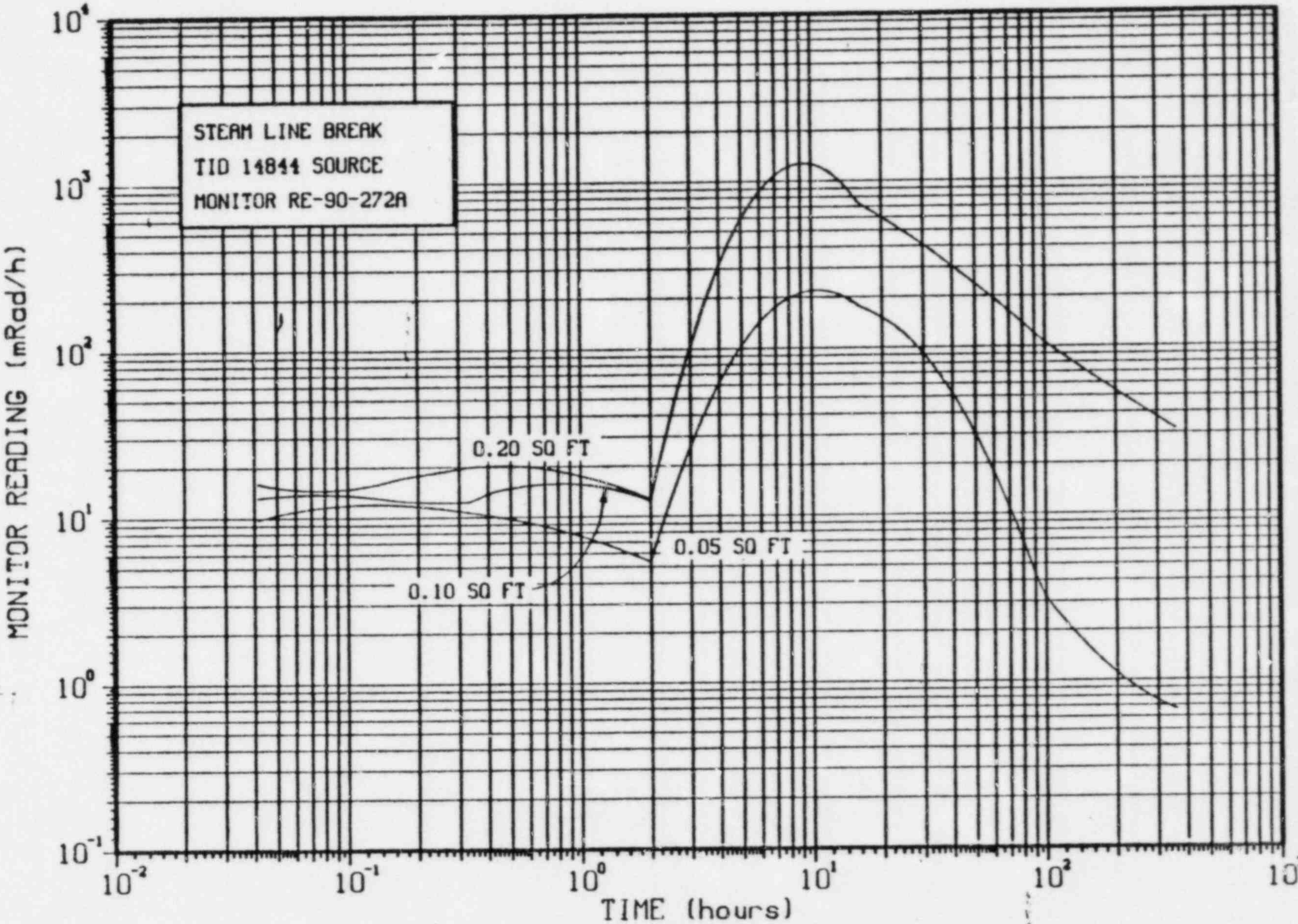
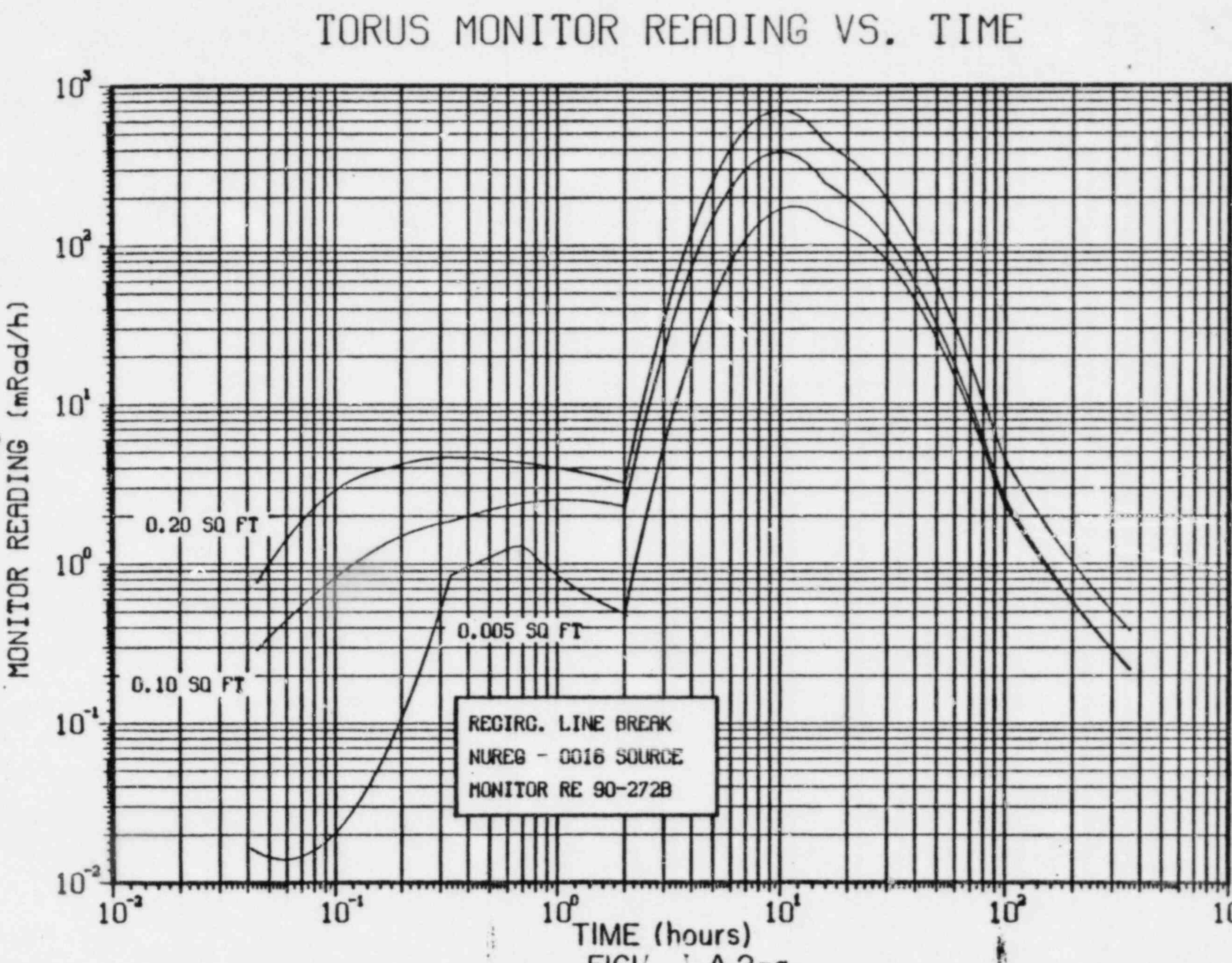


FIGURE A.I-d



TORUS MONITOR READING VS. TIME
MONITOR READING (mRad/h)
TIME (hours)
0.20 SQ FT
0.10 SQ FT
0.005 SQ FT
RECIRC. LINE BREAK
NUREG - 0016 SOURCE
MONITOR RE 90-272B

FIGU : A.2-a

TORUS MONITOR READING VS. TIME

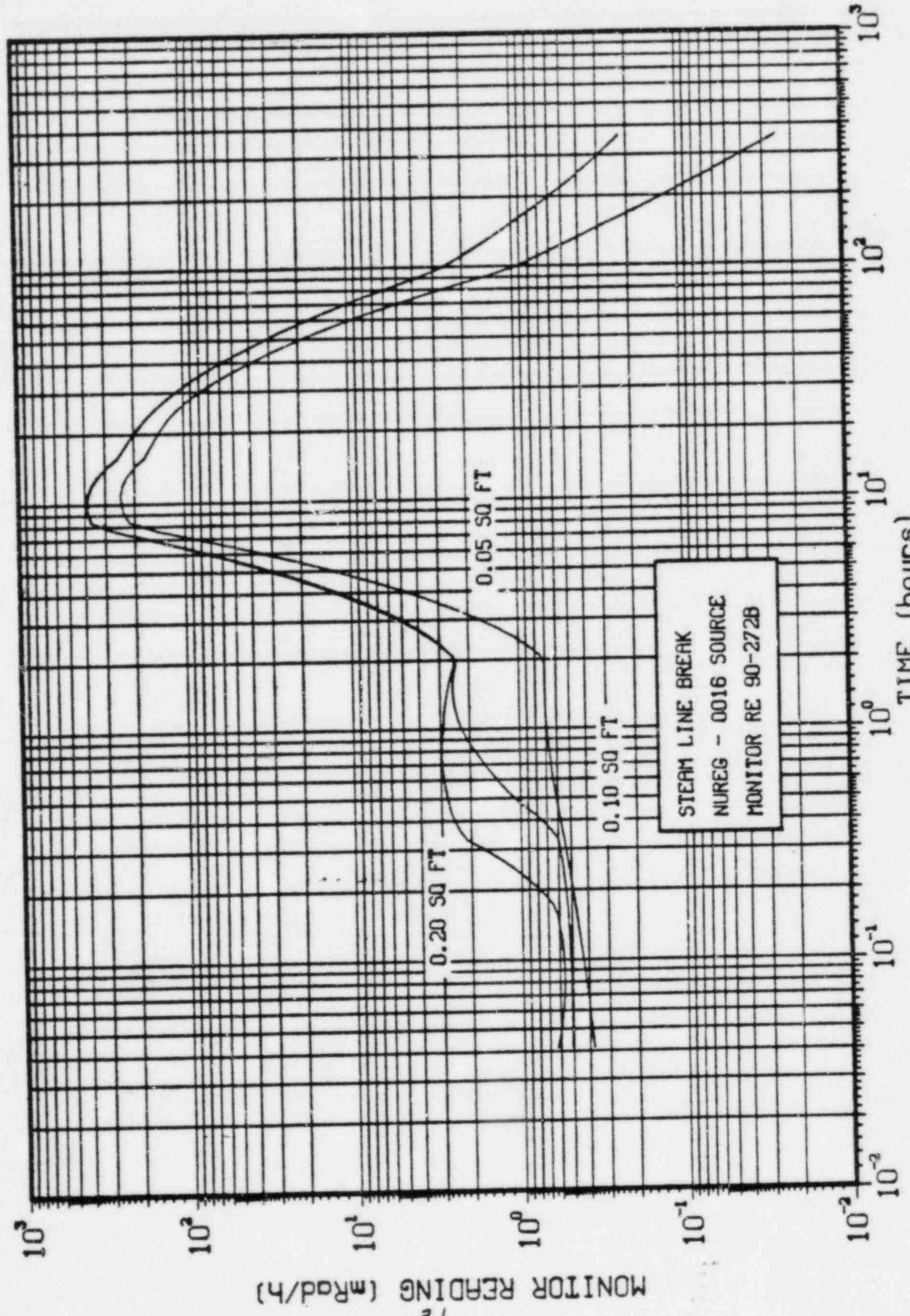


FIGURE A.2-b

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TORUS MONITOR READING VS. TIME

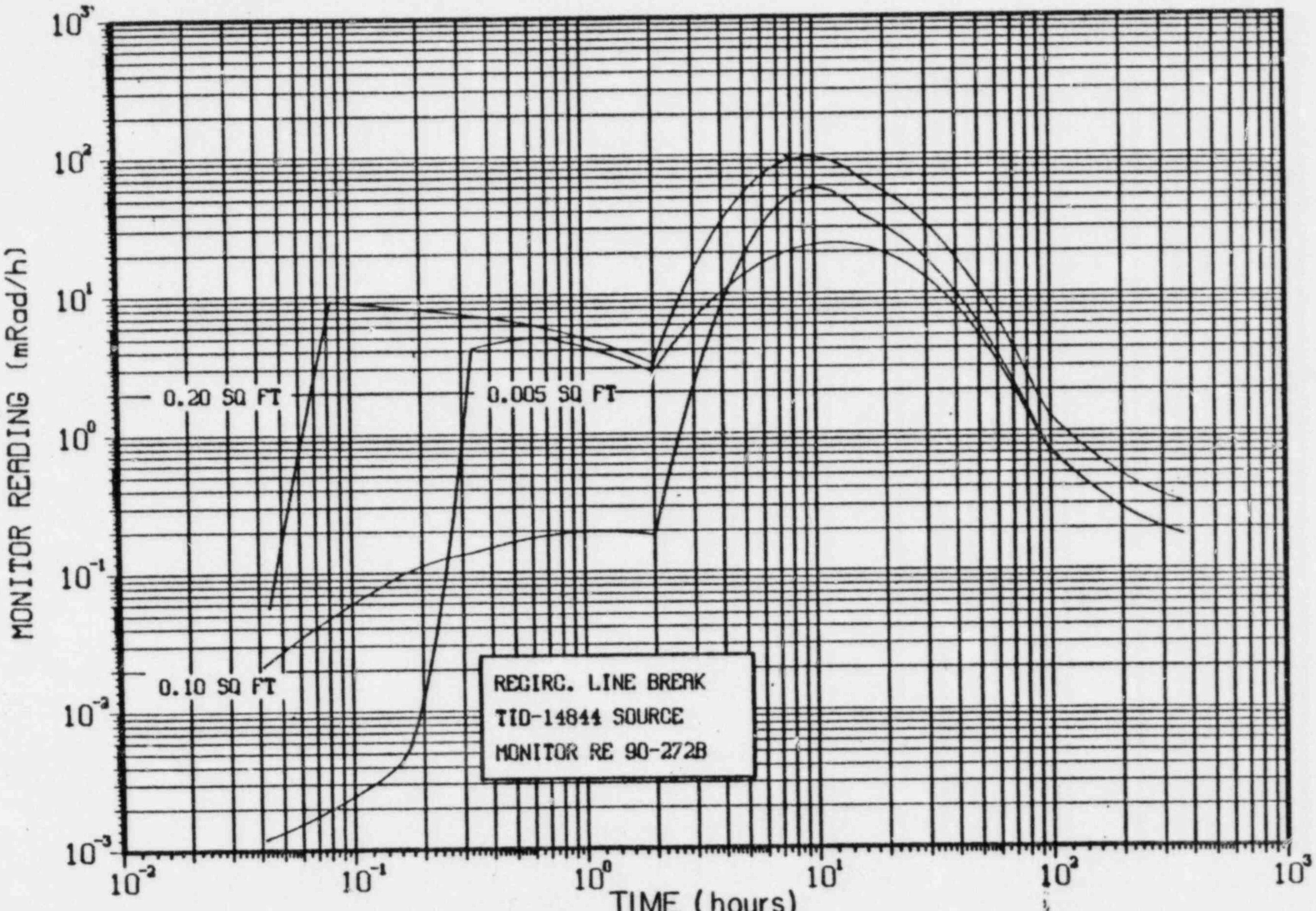
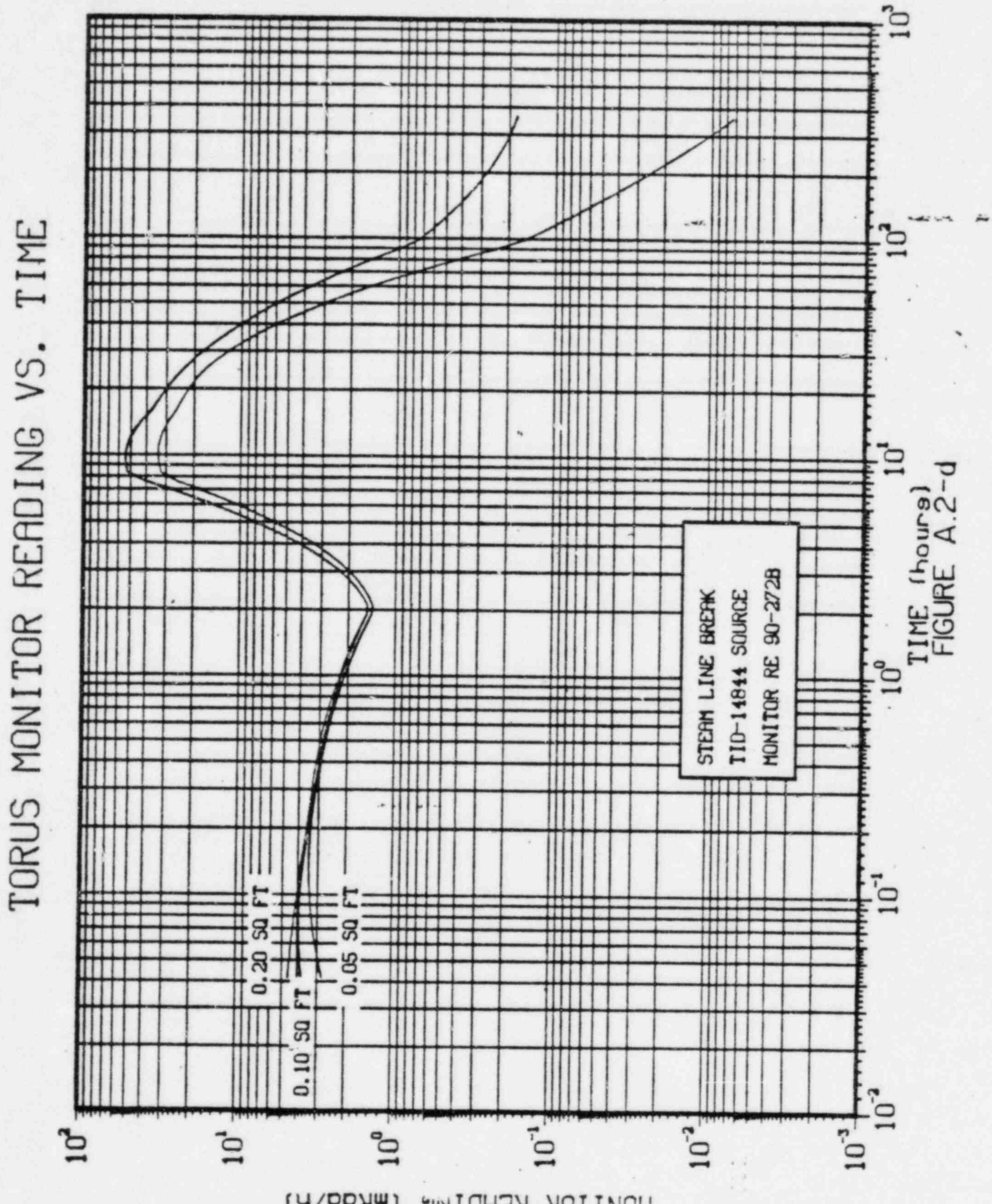


FIGURE A.2-c


TORUS MONITOR READING VS. TIME
STEAM LINE BREAK
TID-14844 SOURCE
MONITOR RE 90-272B
0.20 SQ FT
0.10 SQ FT
0.05 SQ FT
10^{2}
10^{1}
10^{0}
10^{-1}
10^{-2}
10^{-3}
MONITOR READING (mRad/h)
10^{-2}
10^{-1}
10^{0}
10^{1}
10^{2}
10^{3}
TIME (hours)
FIGURE A.2-d

CONTAINMENT LEAK RATE VS. PRESSURE

FOR DIFFERENT HOLE SIZES

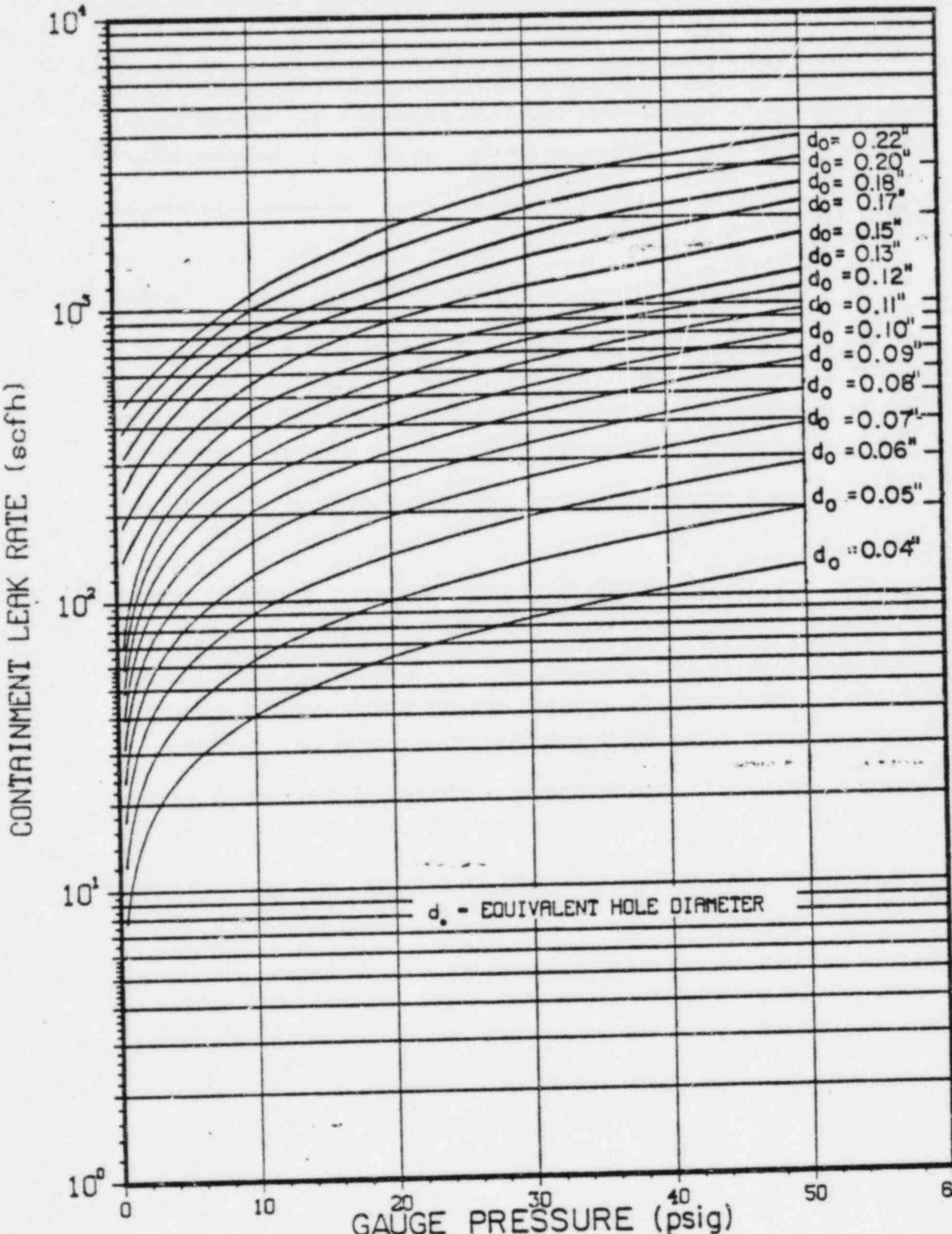


FIGURE A.3

NOBLE GAS RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES

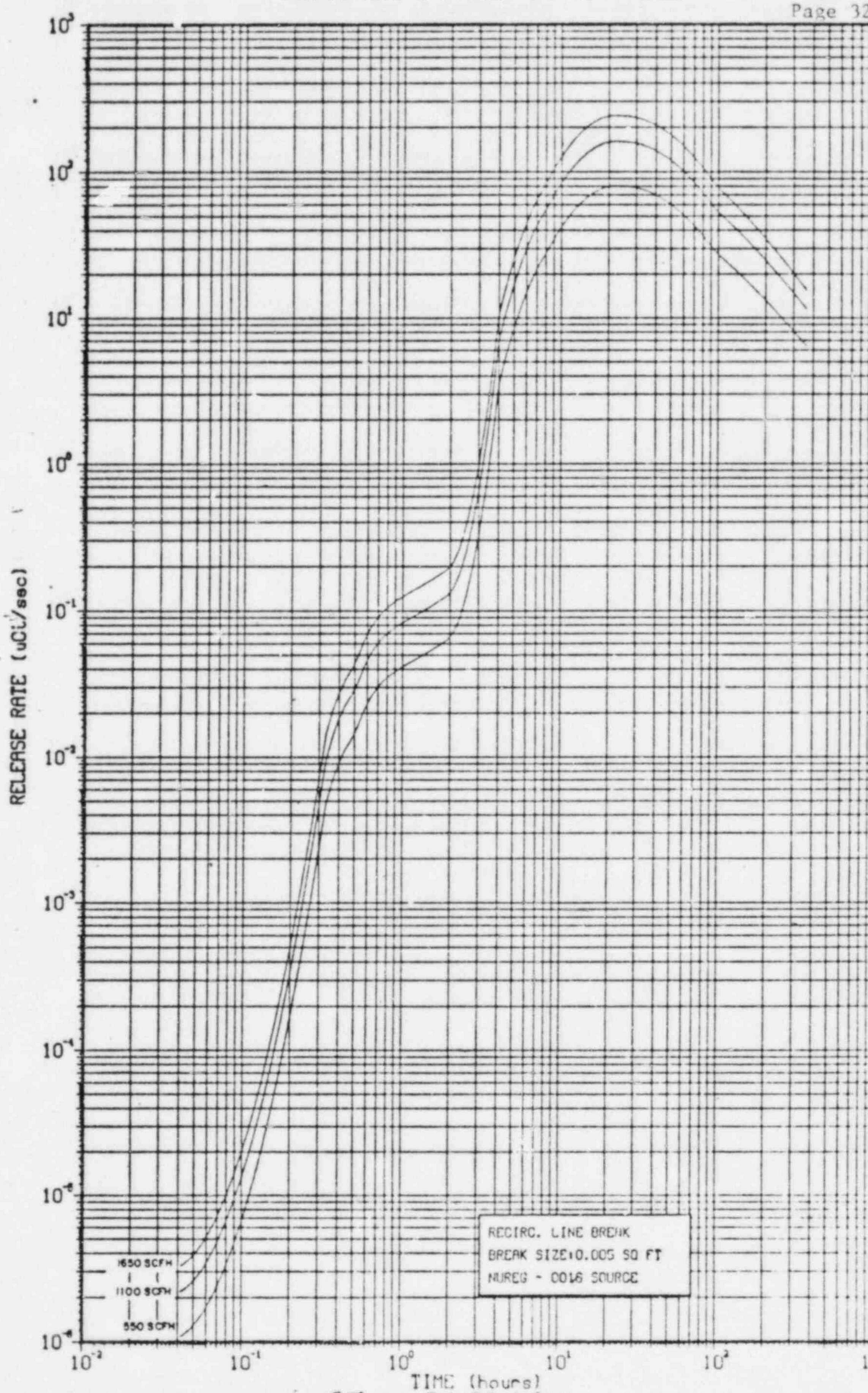


FIGURE A.4-a

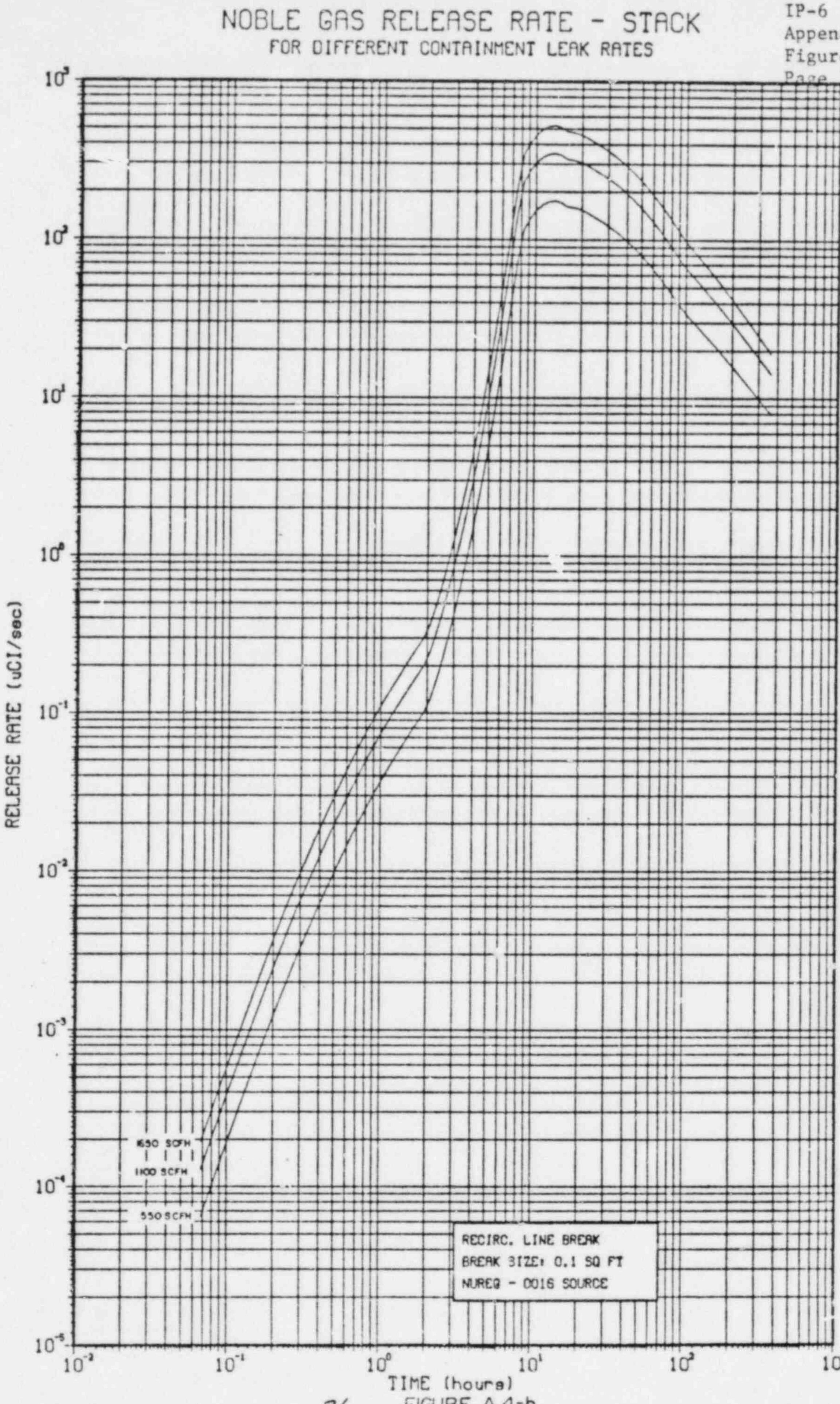

NOBLE GAS RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
RECIRC. LINE BREAK
BREAK SIZE: 0.1 SQ FT
NUREG - 0016 SOURCE

FIGURE A.4-b

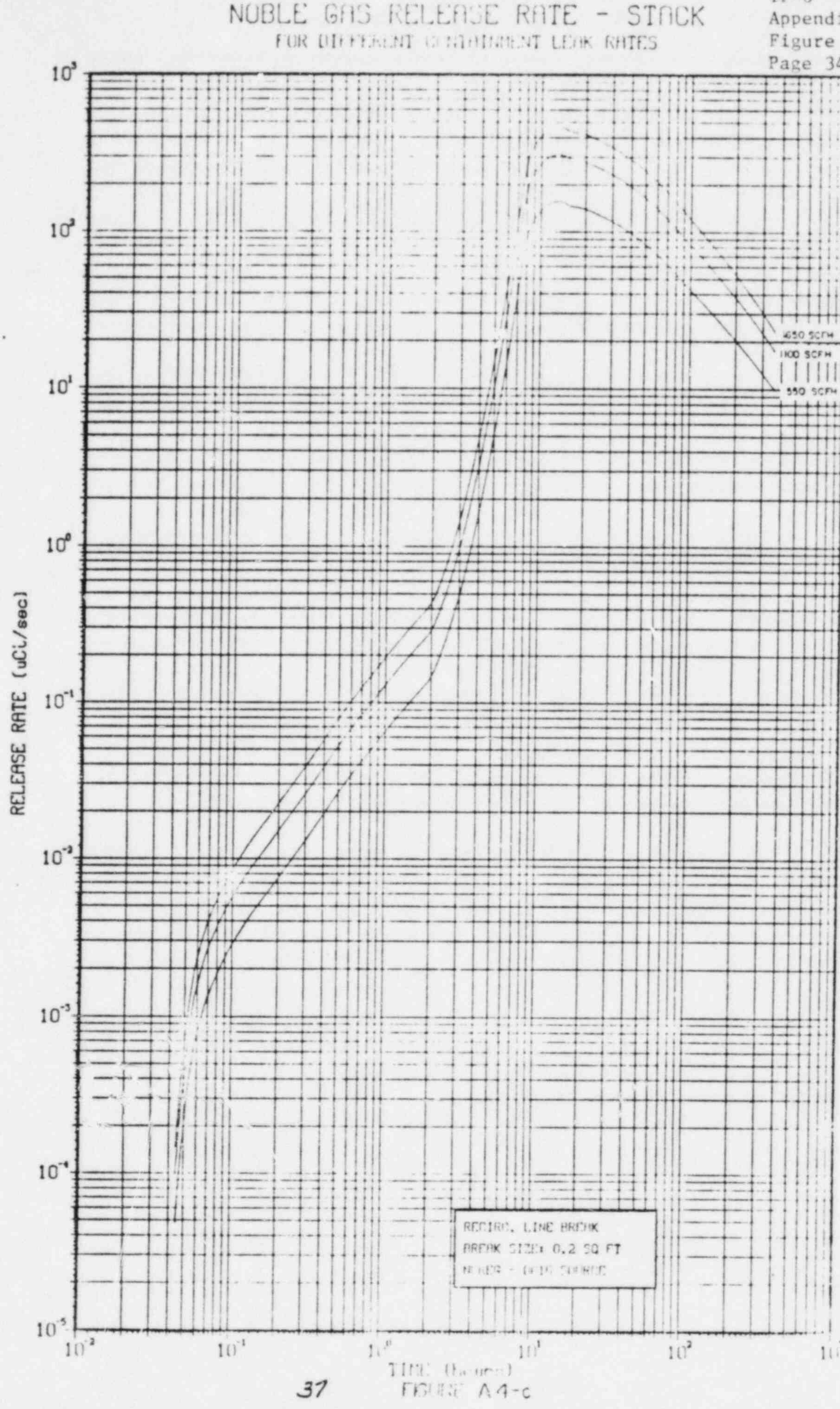

NOBLE GAS RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCi/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
RECIRC. LINE BREAK
BREAK SIZE: 0.2 SQ FT

FIGURE A.4-c

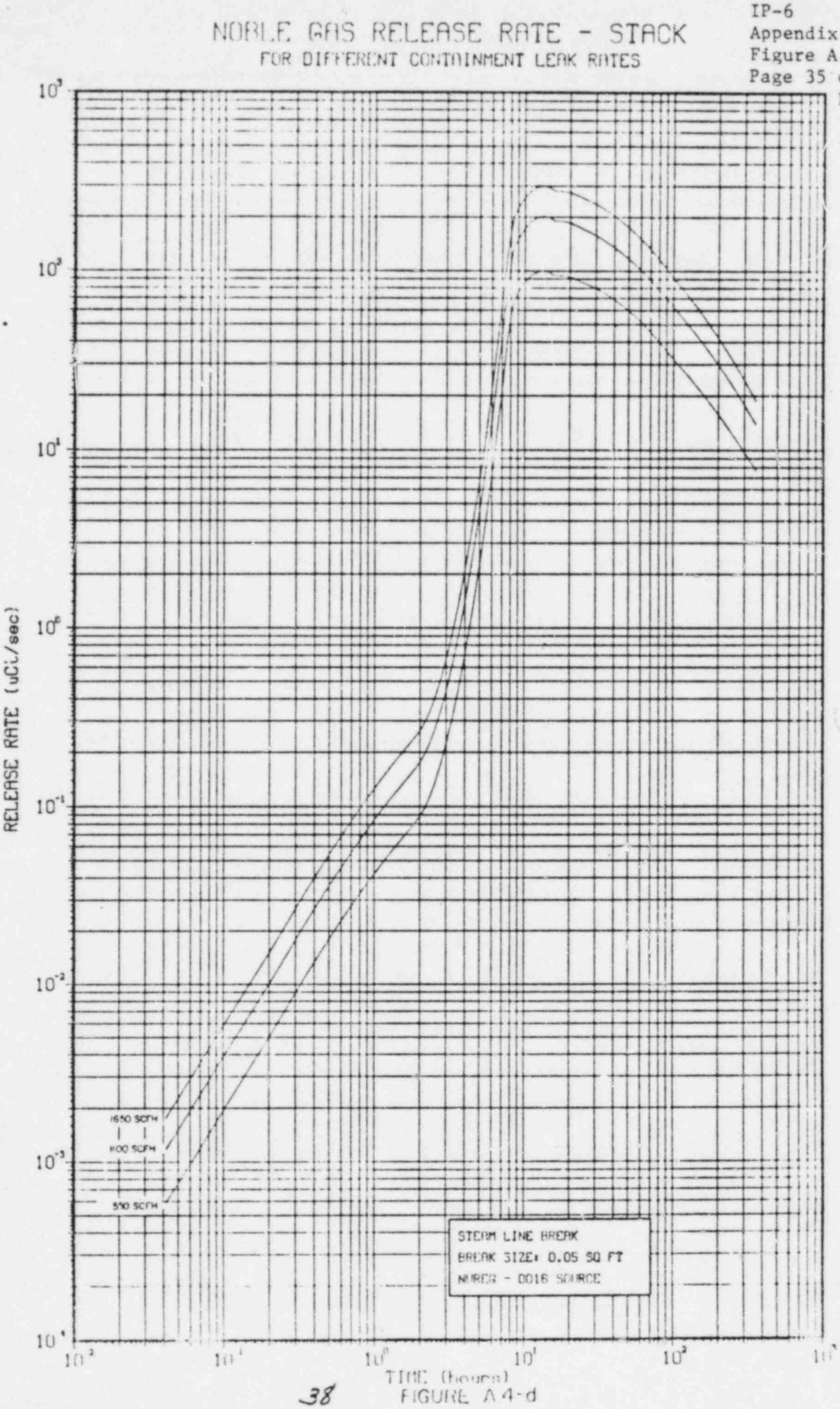

NOBLE GAS RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCi/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
STEAM LINE BREAK
BREAK SIZE: 0.05 SQ FT
NUREG - 0016 SOURCE

FIGURE A.4-d

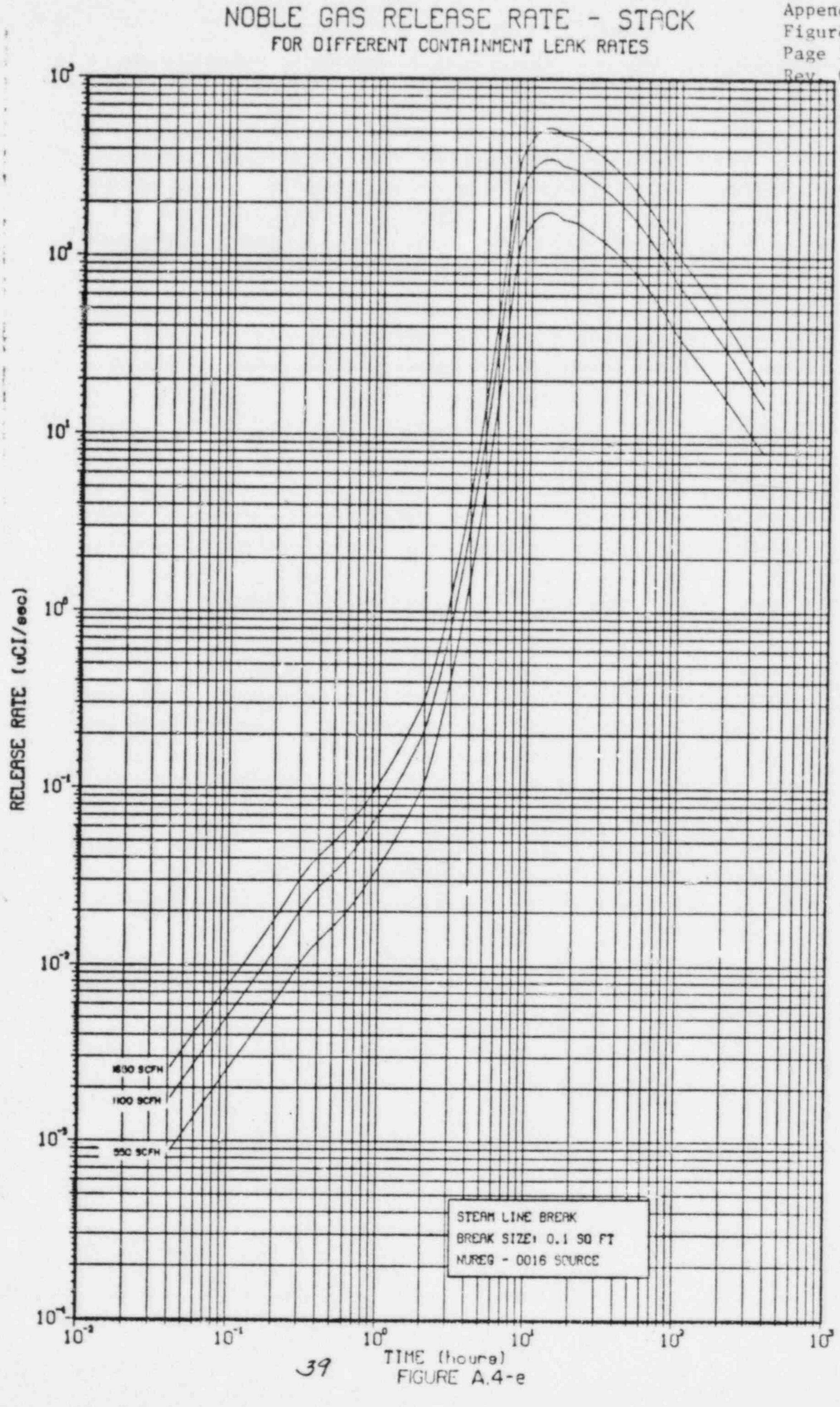
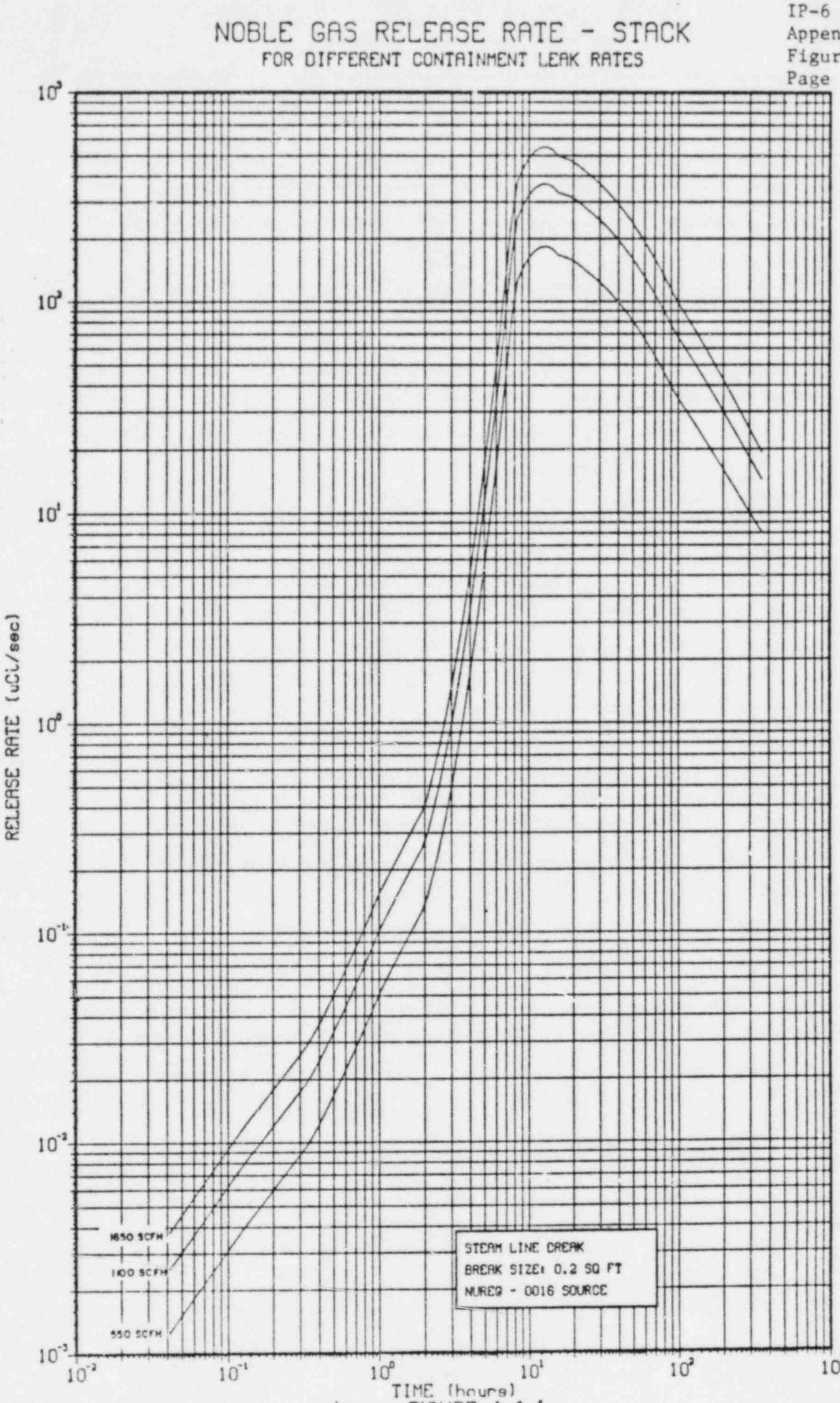

NOBLE GAS RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
1600 SCFH
1100 SCFH
550 SCFH
STEAM LINE BREAK
BREAK SIZE: 0.1 SQ FT
NUREG - 0016 SOURCE
TIME (hours)

FIGURE A.4-e



40 FIGURE A.4-f

NOBLE GAS RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES

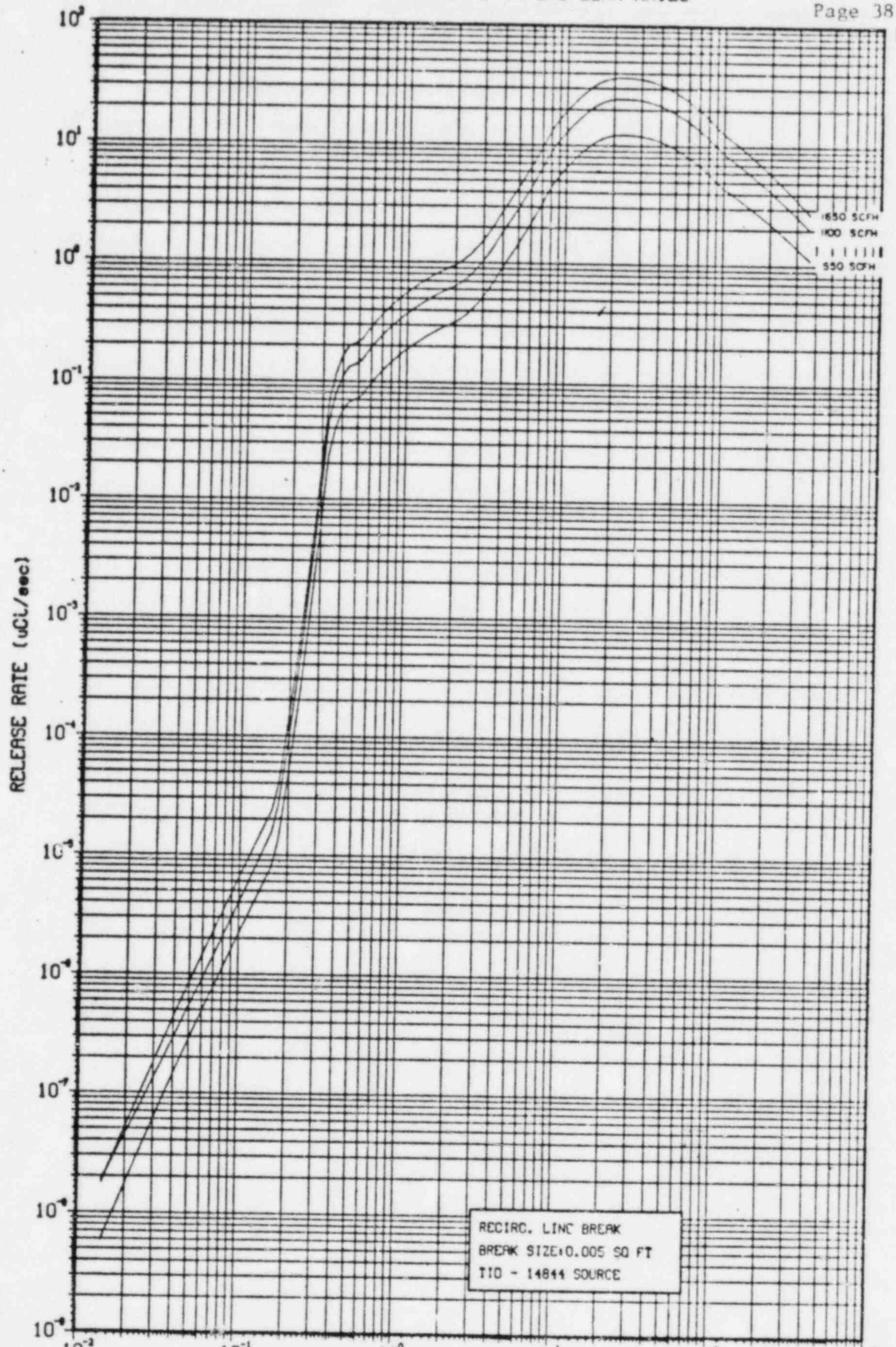


FIGURE A.4-g

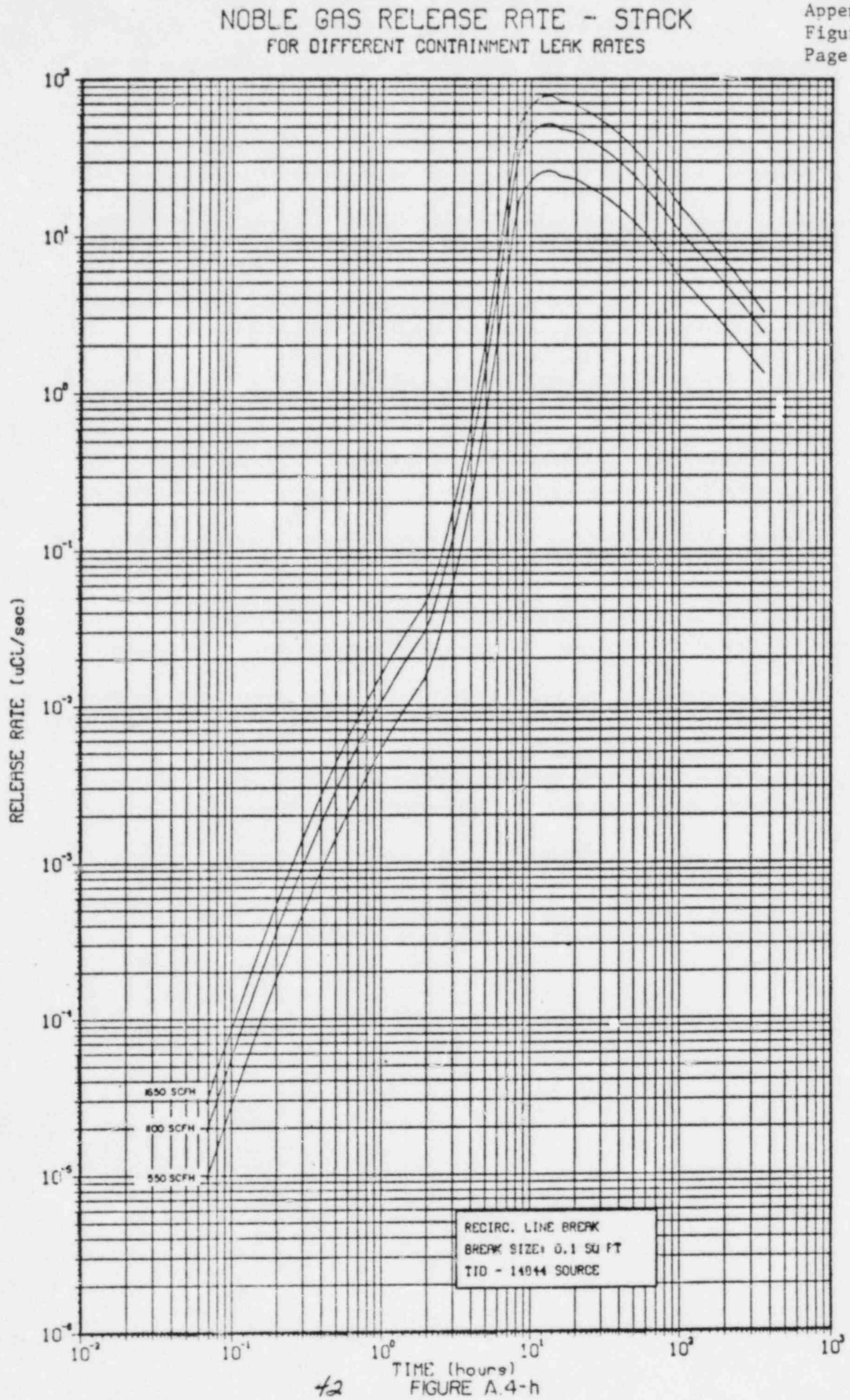


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FIGURE A.4-h

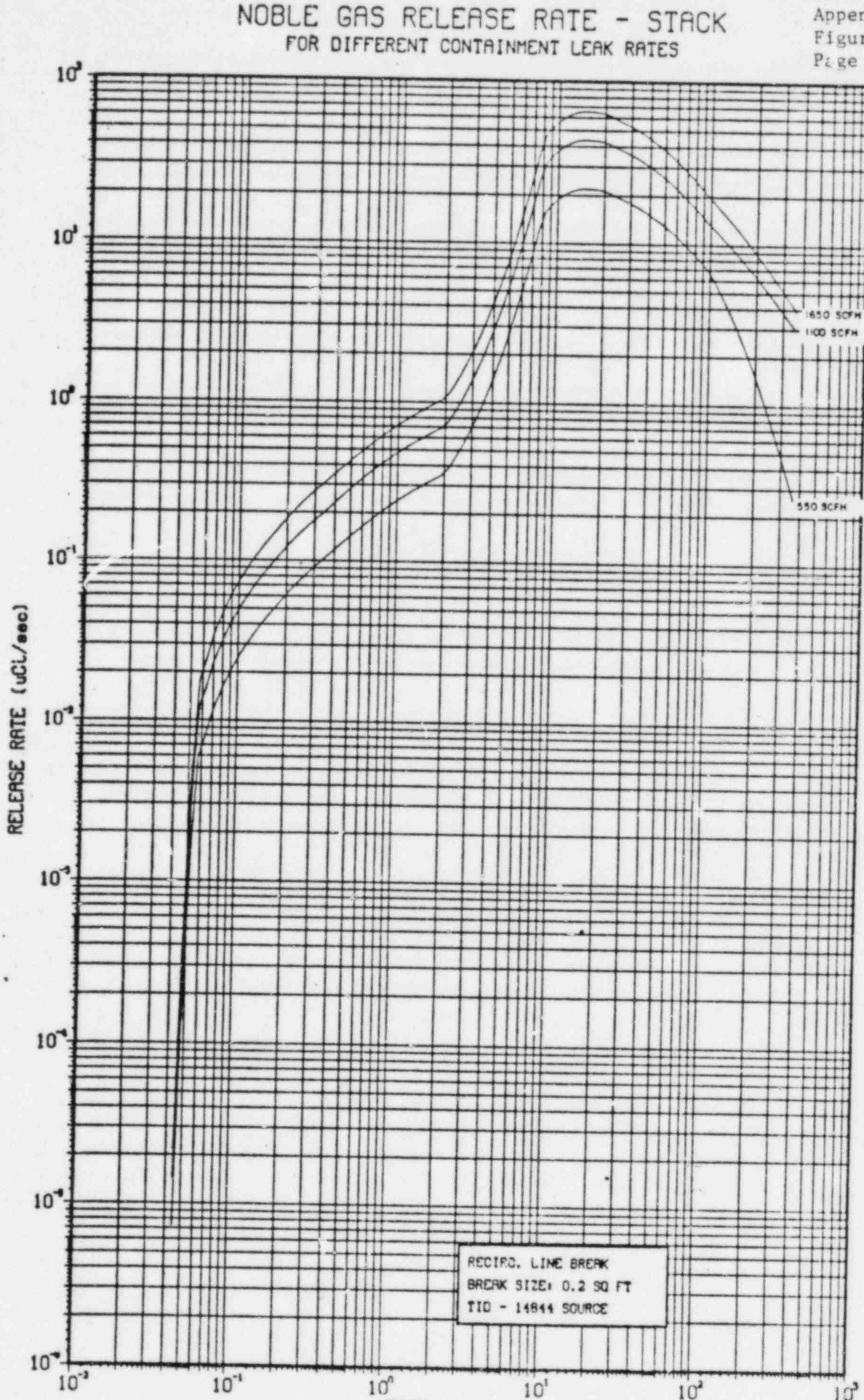
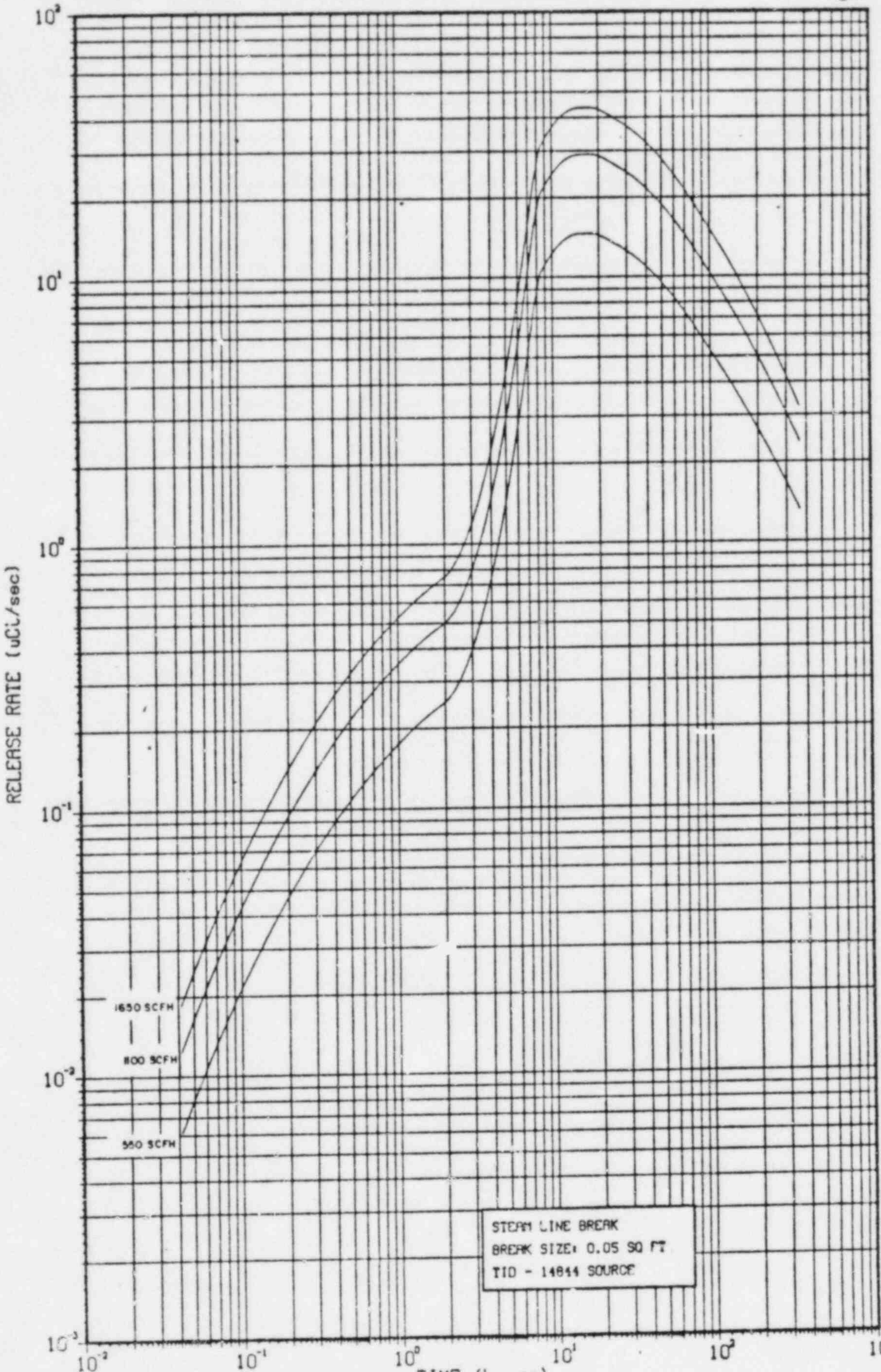

NOBLE GAS RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCi/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
RECIRC. LINE BREAK
BREAK SIZE: 0.2 SQ FT
TID - 14844 SOURCE

FIGURE A.4-1

NOBLE GAS RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES



44 FIGURE A.4-j

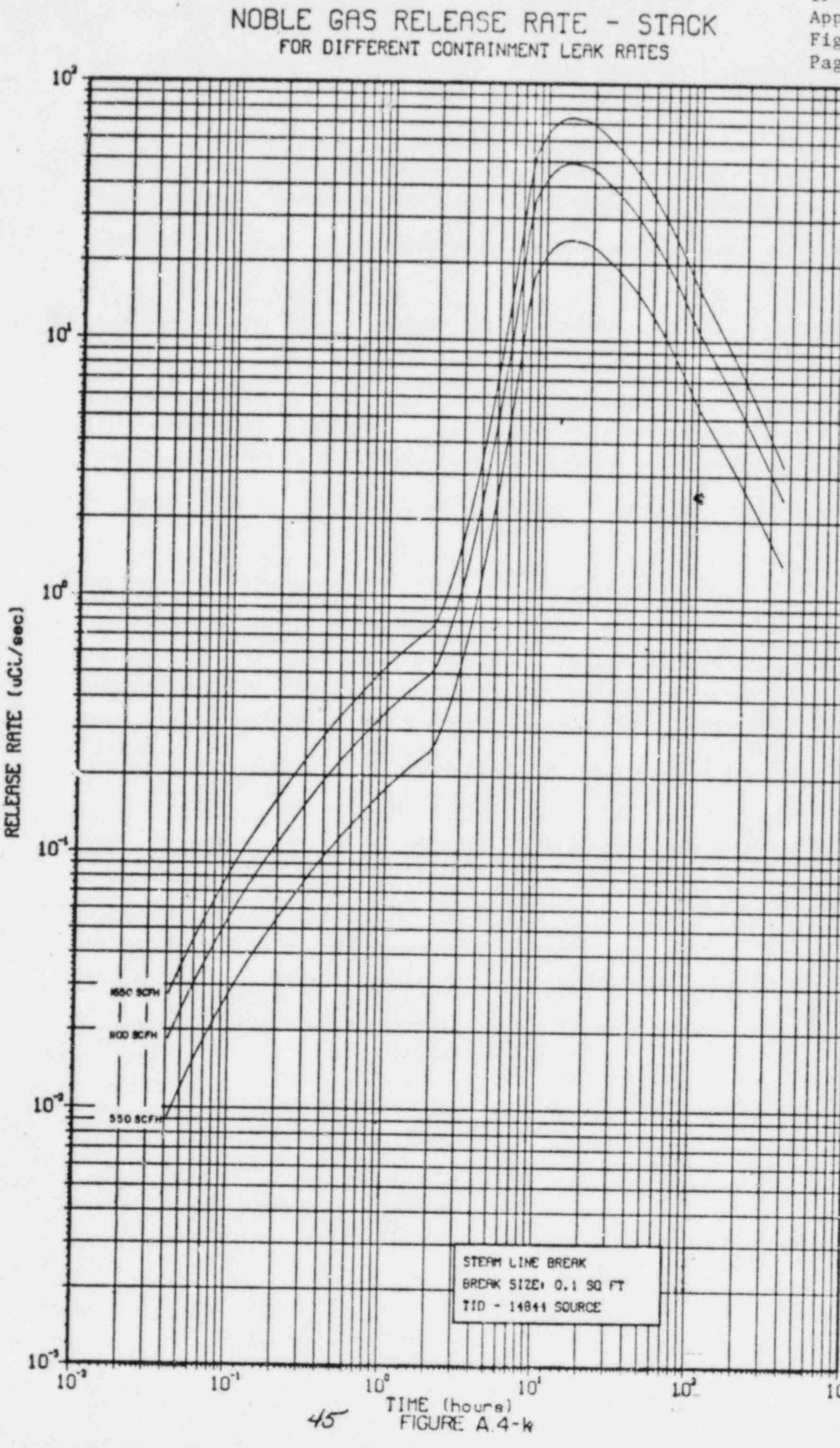

NOBLE GAS RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCi/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
STEAM LINE BREAK
BREAK SIZE: 0.1 SQ FT
TID - 14844 SOURCE

FIGURE A.4-k

NOBLE GAS RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES

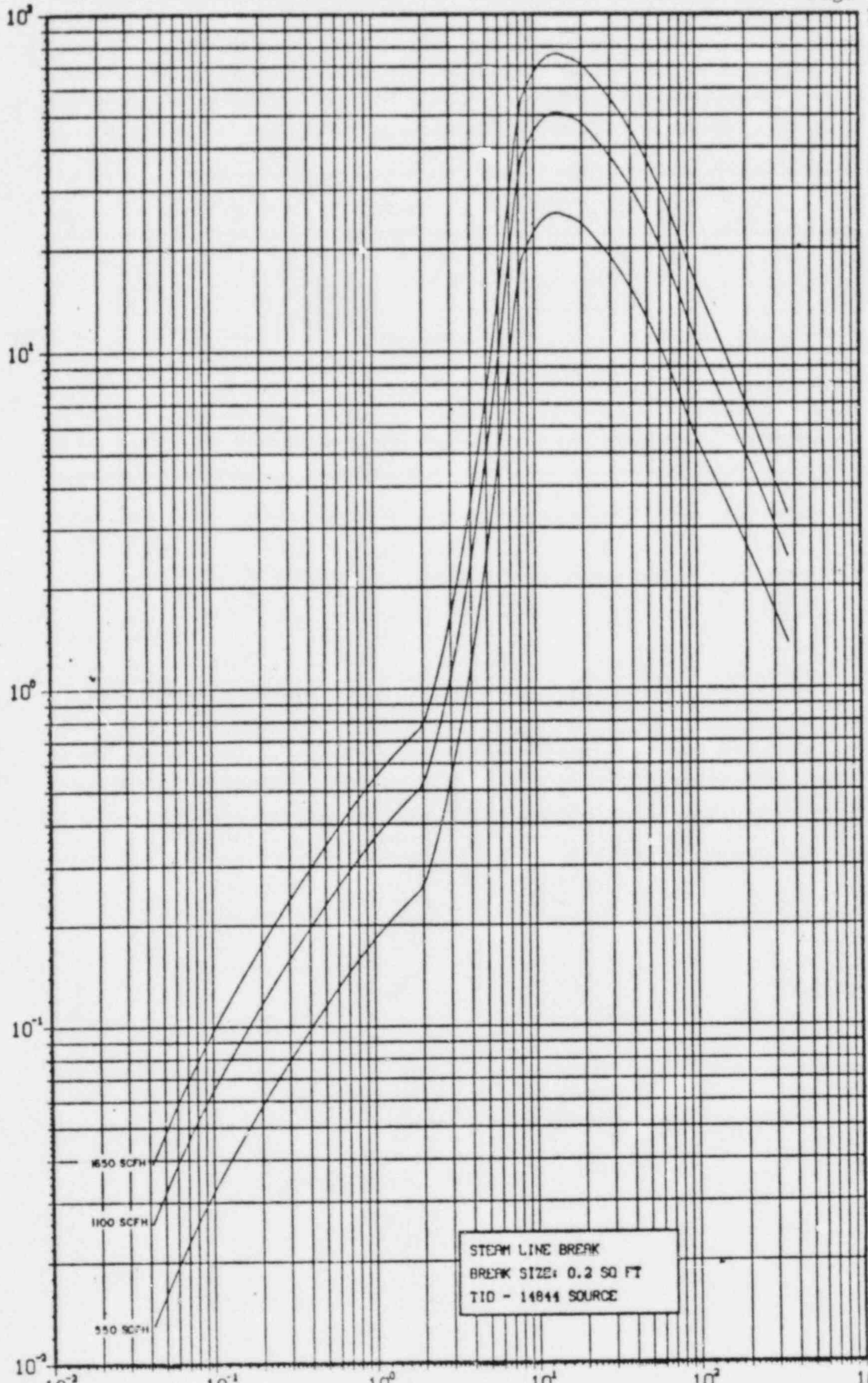


FIGURE A.4-1

IODINE RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES

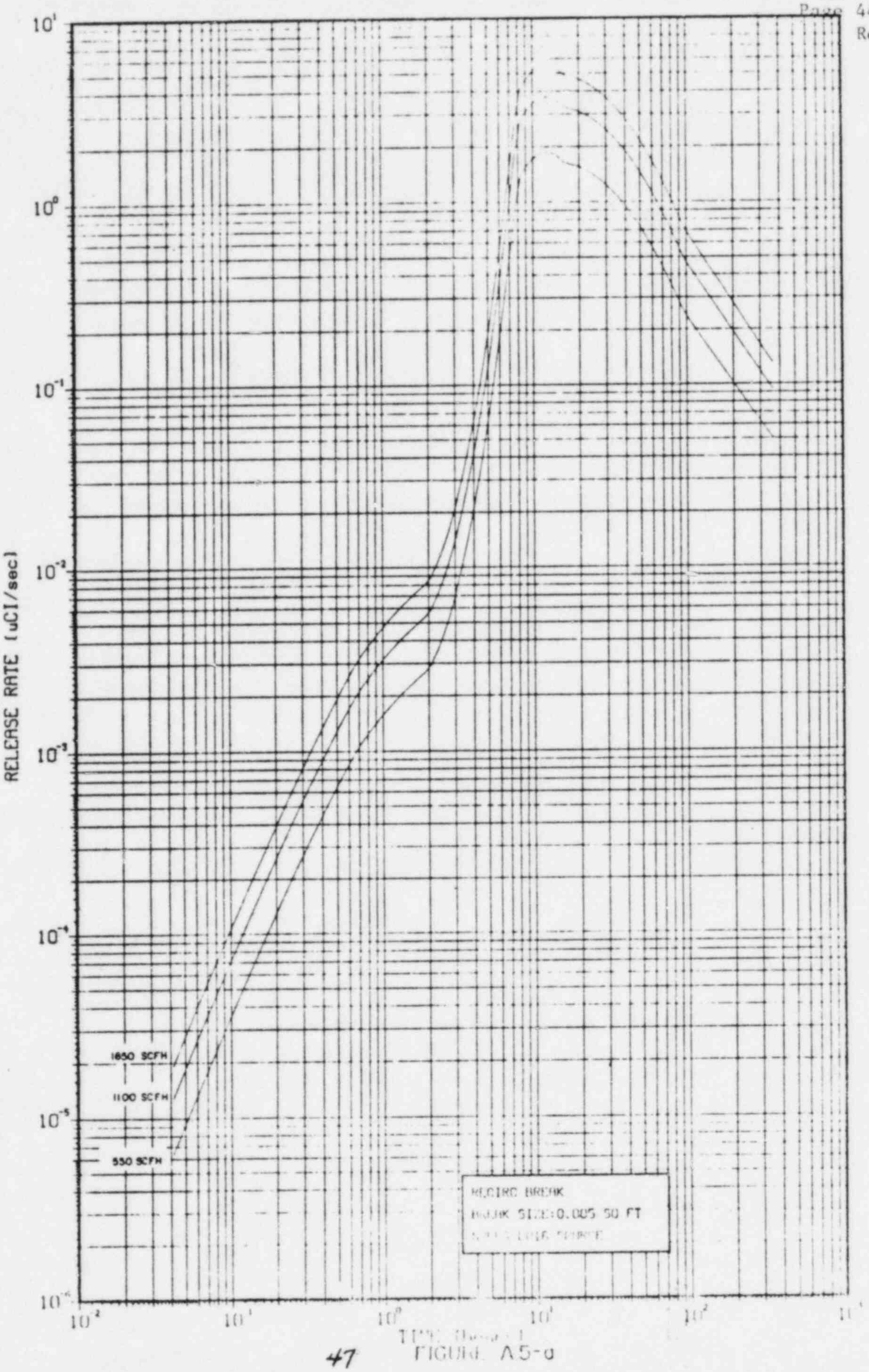


FIGURE A5-a

IODINE RELEASE RATE - STACK

FOR DIFFERENT CONTAINMENT LEAK RATES

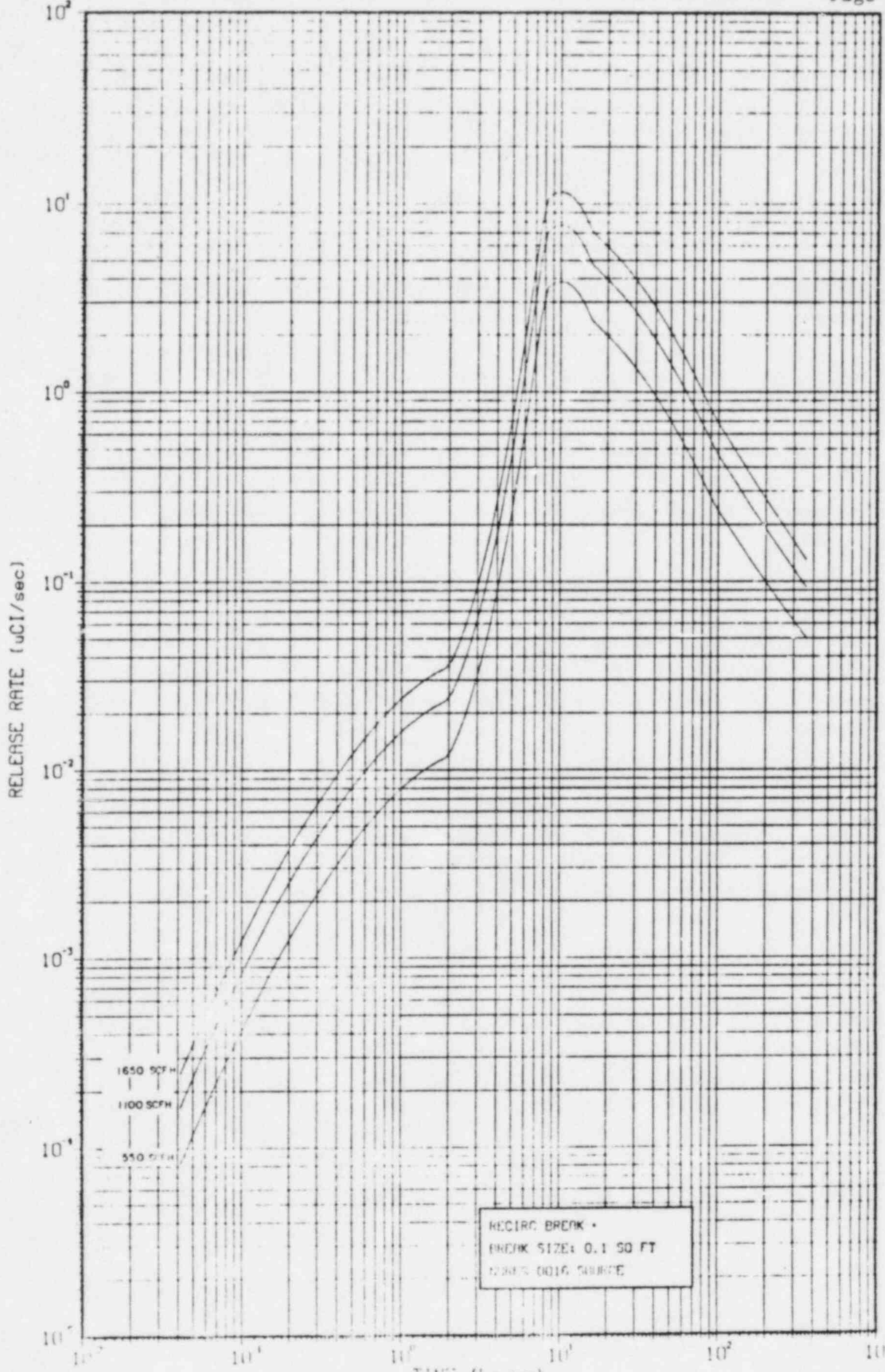


FIGURE A5-b

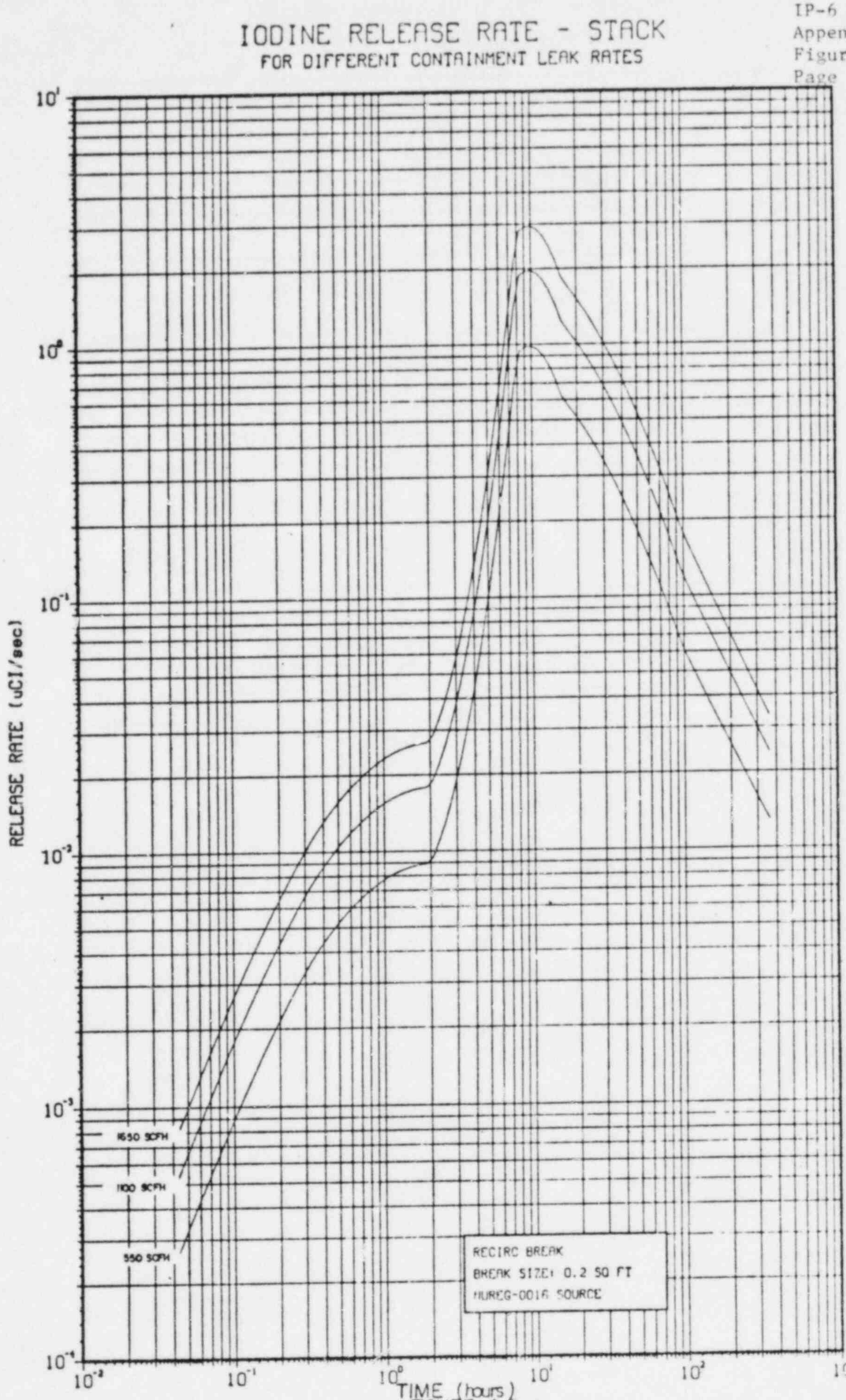

IODINE RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
RECIRC BREAK
BREAK SIZE: 0.2 SQ FT
NUREG-0016 SOURCE

FIGURE A.5-c

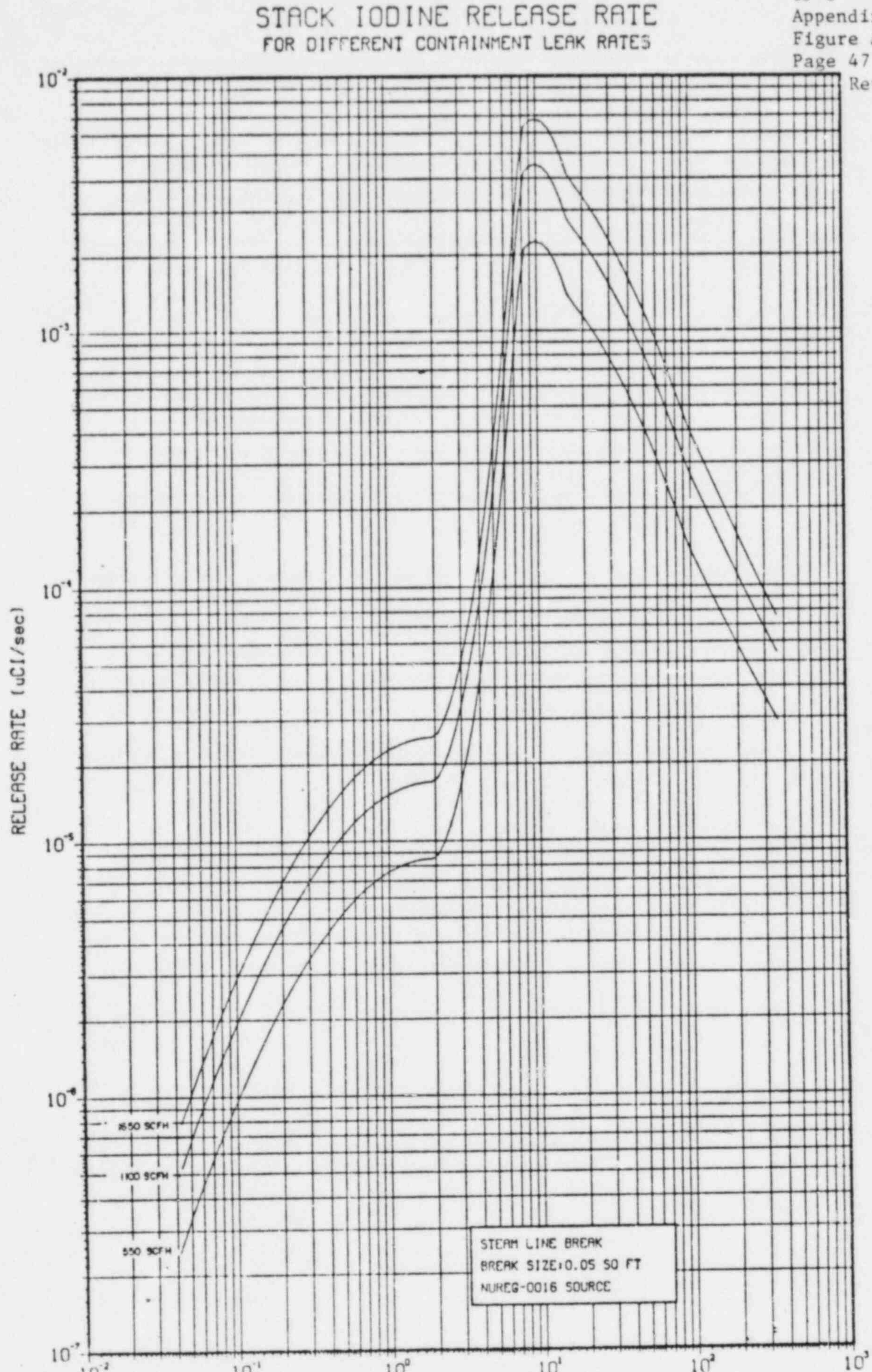

STACK IODINE RELEASE RATE
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
STEAM LINE BREAK
BREAK SIZE: 0.05 SQ FT
NUREG-0016 SOURCE

FIGURE A.5-d

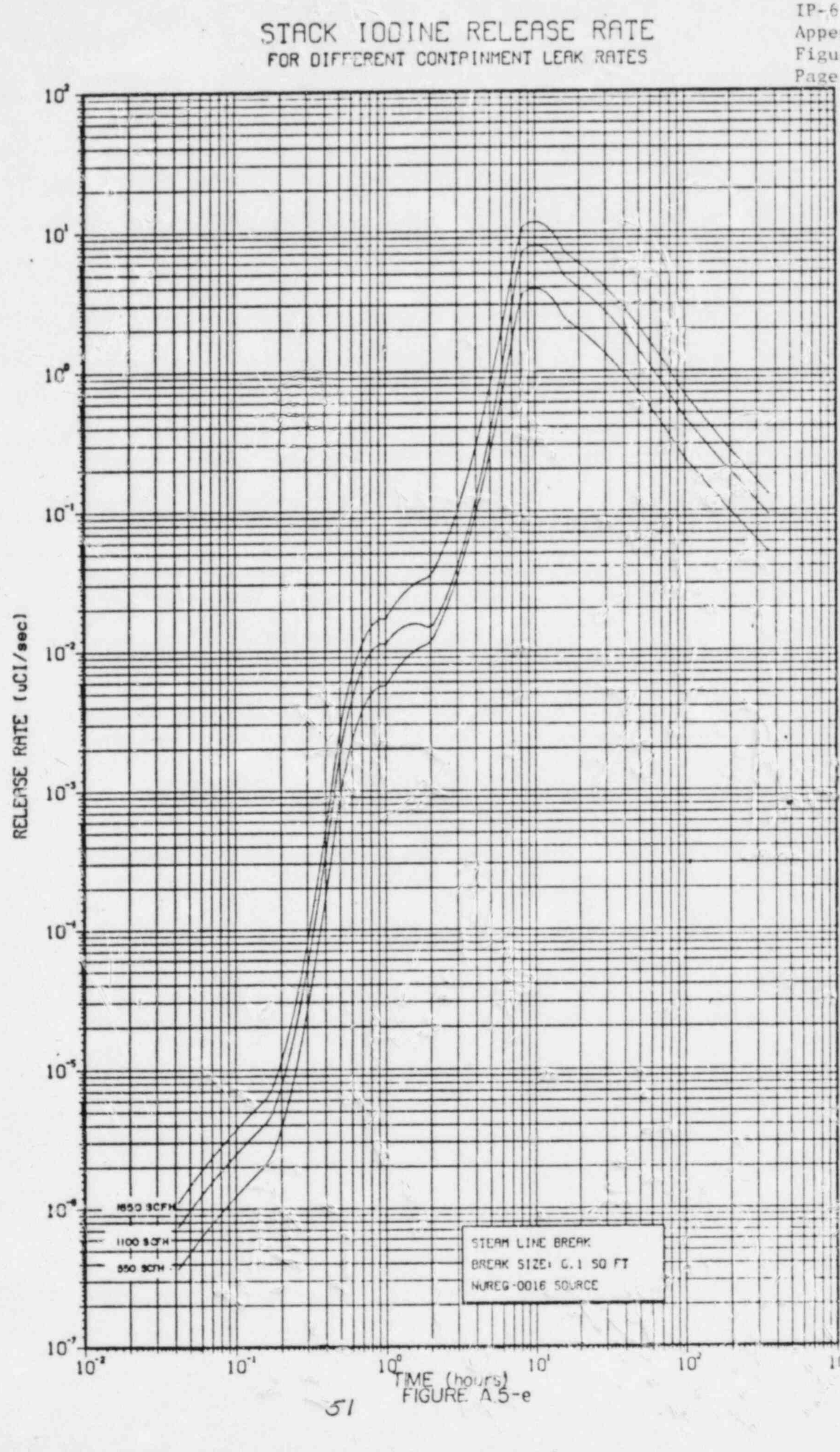

STACK IODINE RELEASE RATE
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
STEAM LINE BREAK
BREAK SIZE: 0.1 SQ FT
NUREG-0016 SOURCE

FIGURE A.5-e

STACK IODINE RELEASE RATE
FOR DIFFERENT CONTAINMENT LEAK RATES

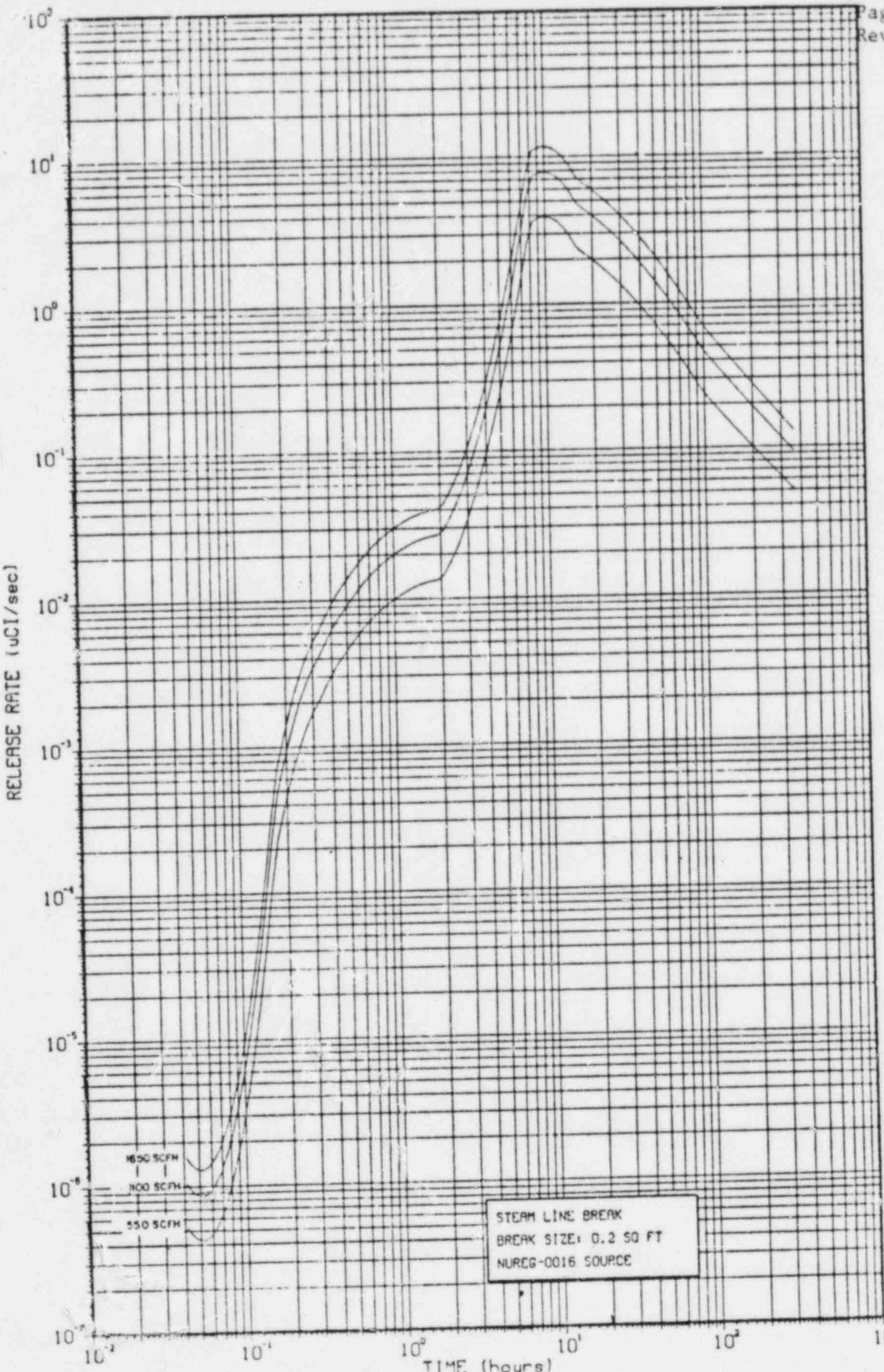


FIGURE A5-f

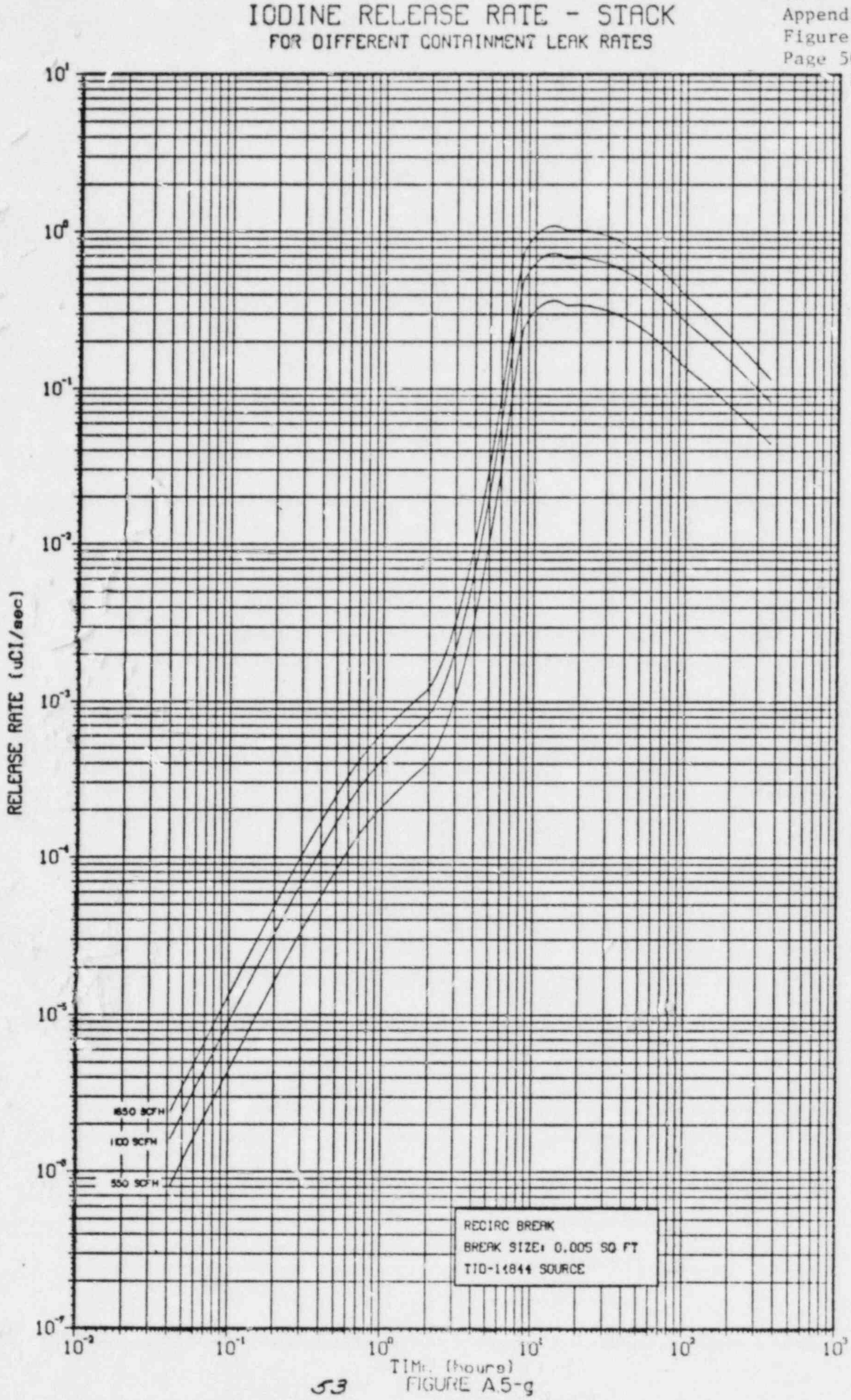
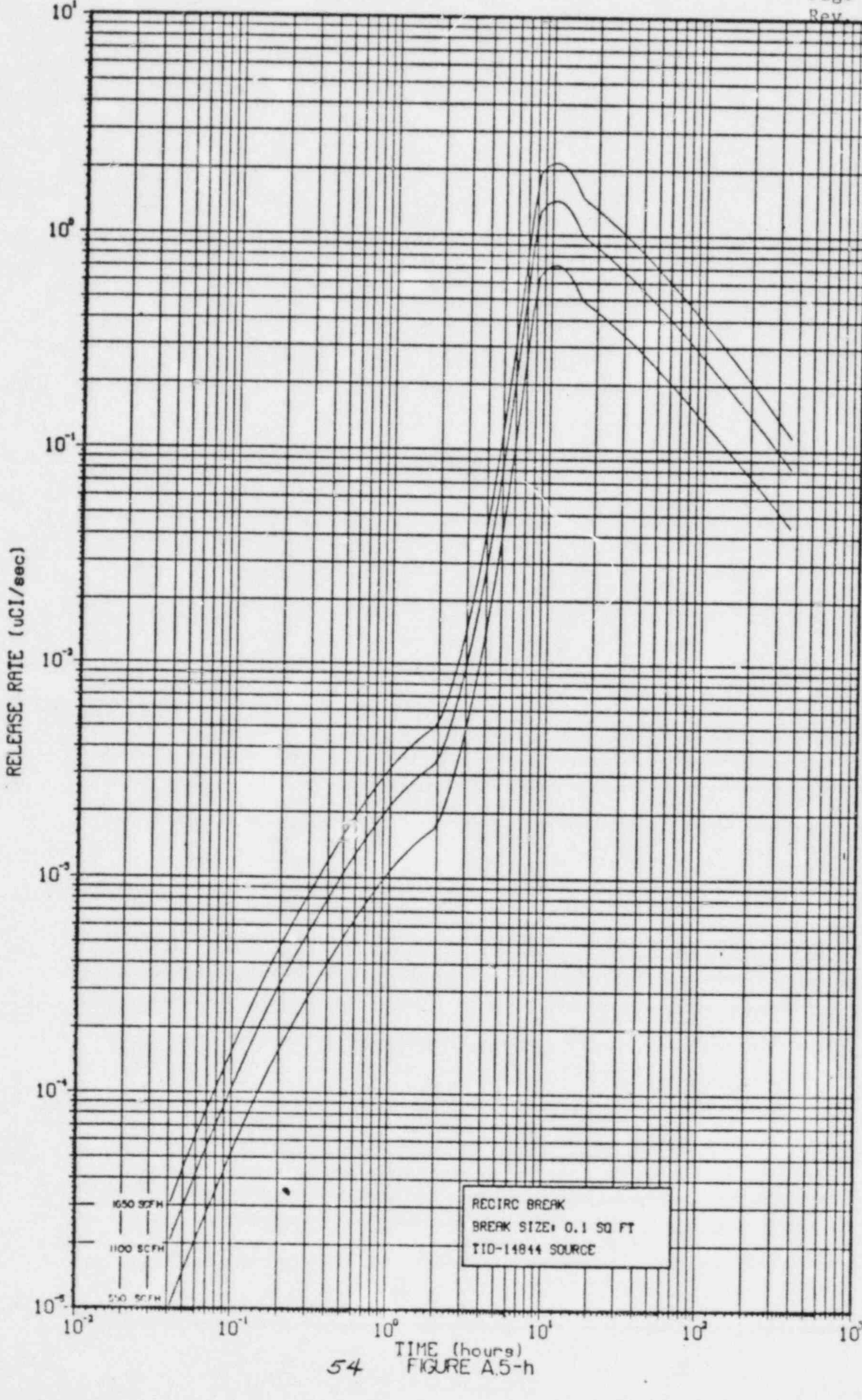

IODINE RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
RECIRC BREAK
BREAK SIZE: 0.005 SQ FT
TID-14844 SOURCE

FIGURE A.5-g

IODINE RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES



54 FIGURE A.5-h

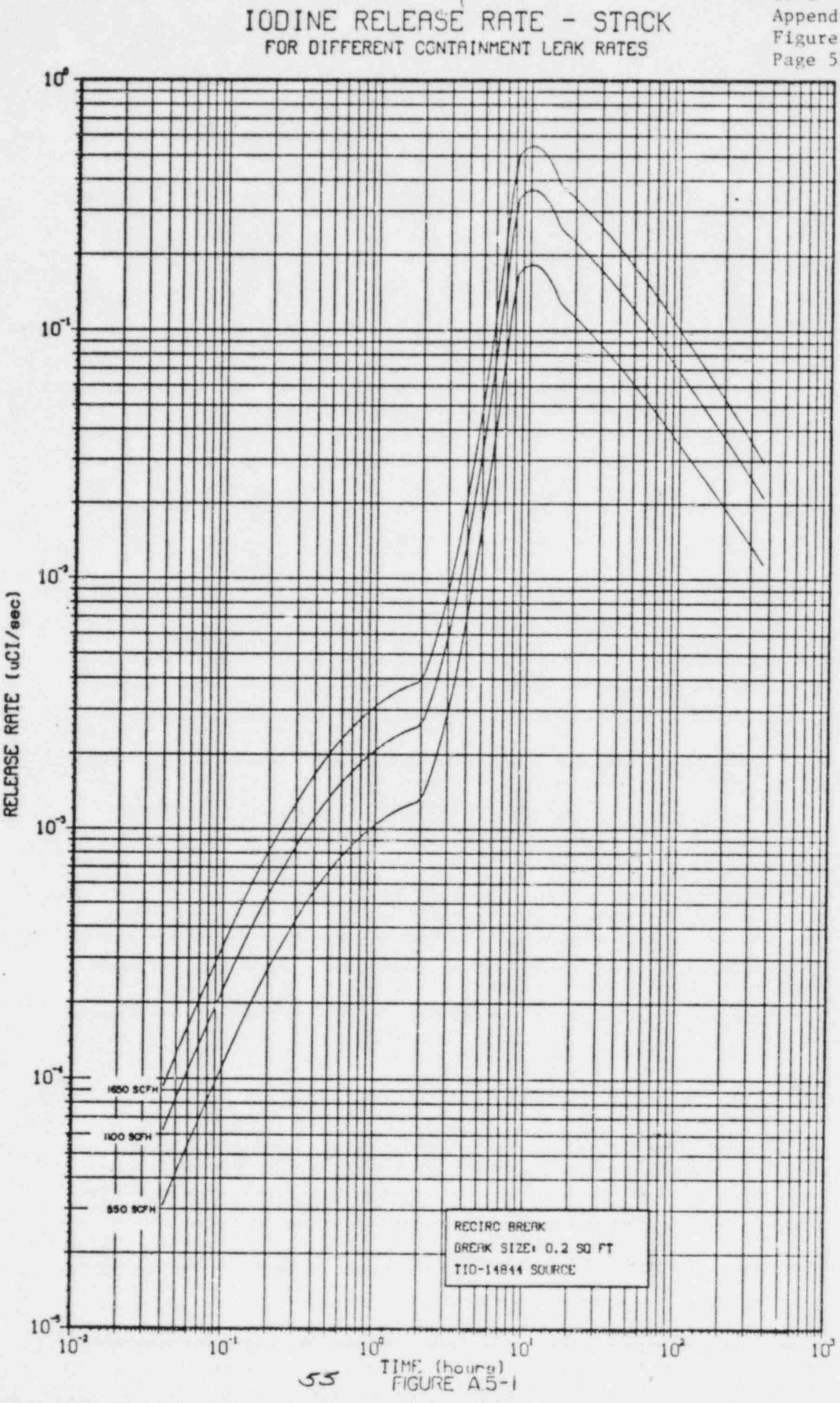

IODINE RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
RECIRC BREAK
BREAK SIZE: 0.2 SQ FT
TID-14844 SOURCE

FIGURE A.5-1

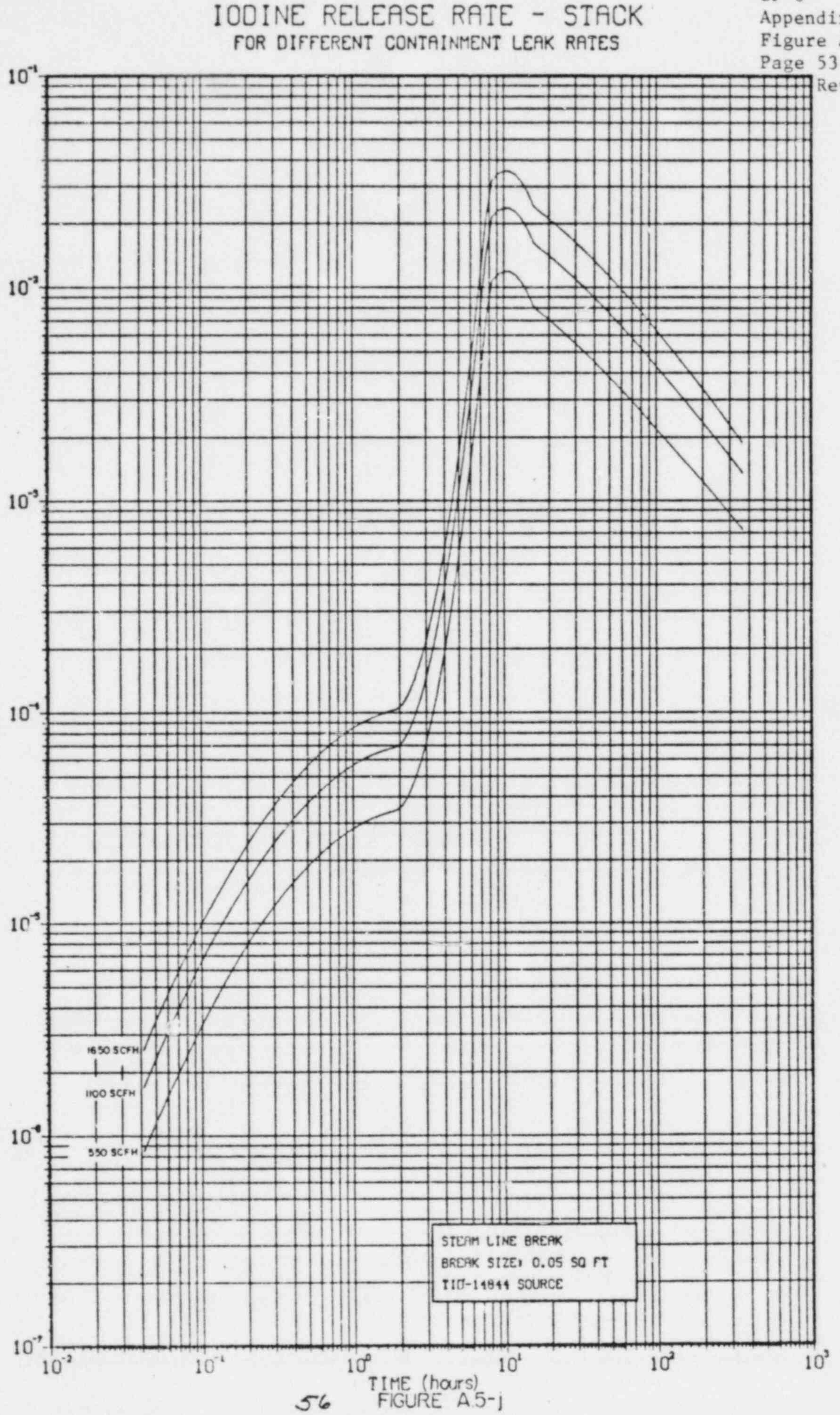
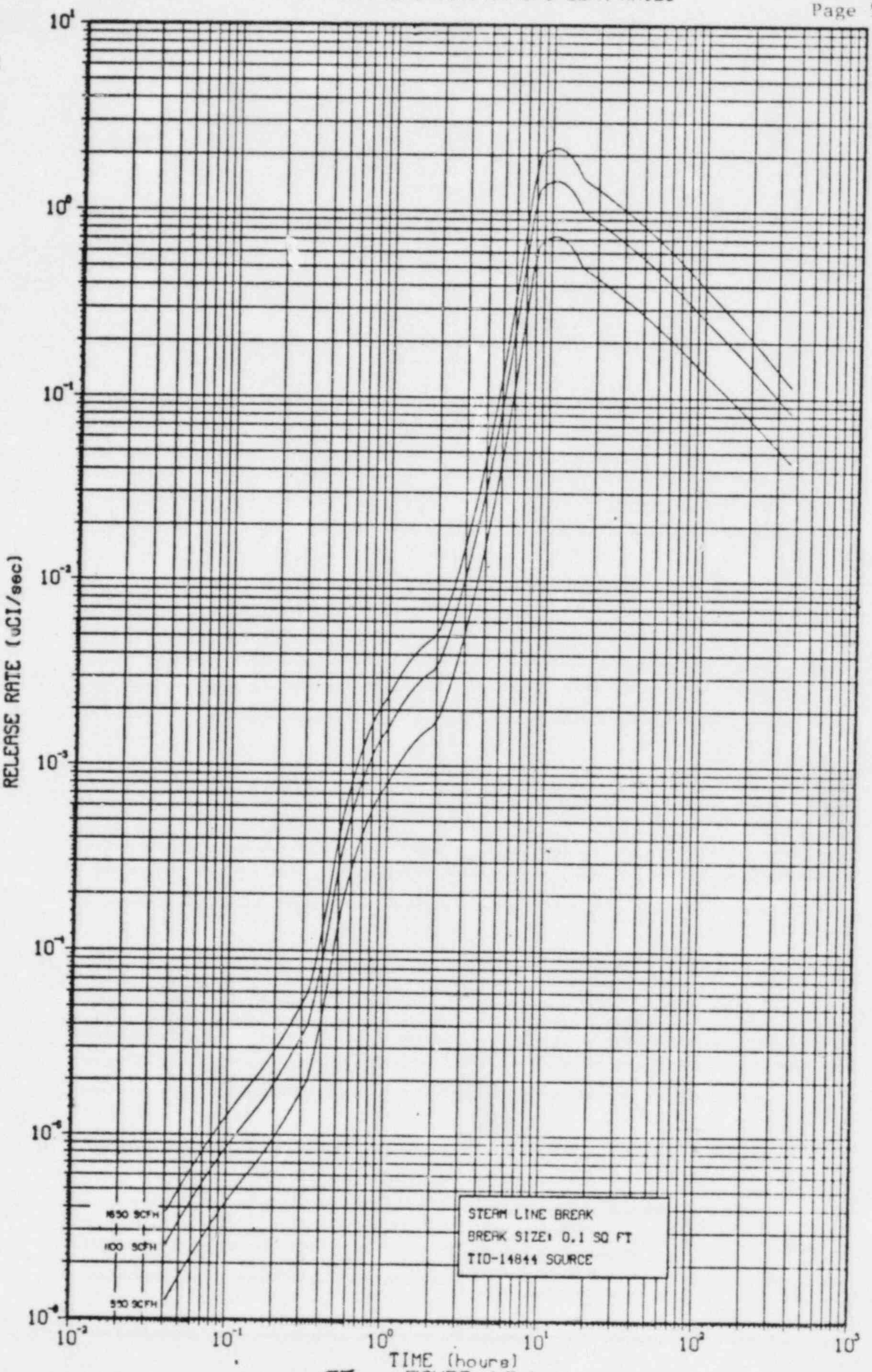

IODINE RELEASE RATE - STACK
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
STEAM LINE BREAK
BREAK SIZE: 0.05 SQ FT
TID-14844 SOURCE

FIGURE A.5-j

IODINE RELEASE RATE - STACK

FOR DIFFERENT CONTAINMENT LEAK RATES



57 FIGURE A.5-k

IODINE RELEASE RATE - STACK

FOR DIFFERENT CONTAINMENT LEAK RATES

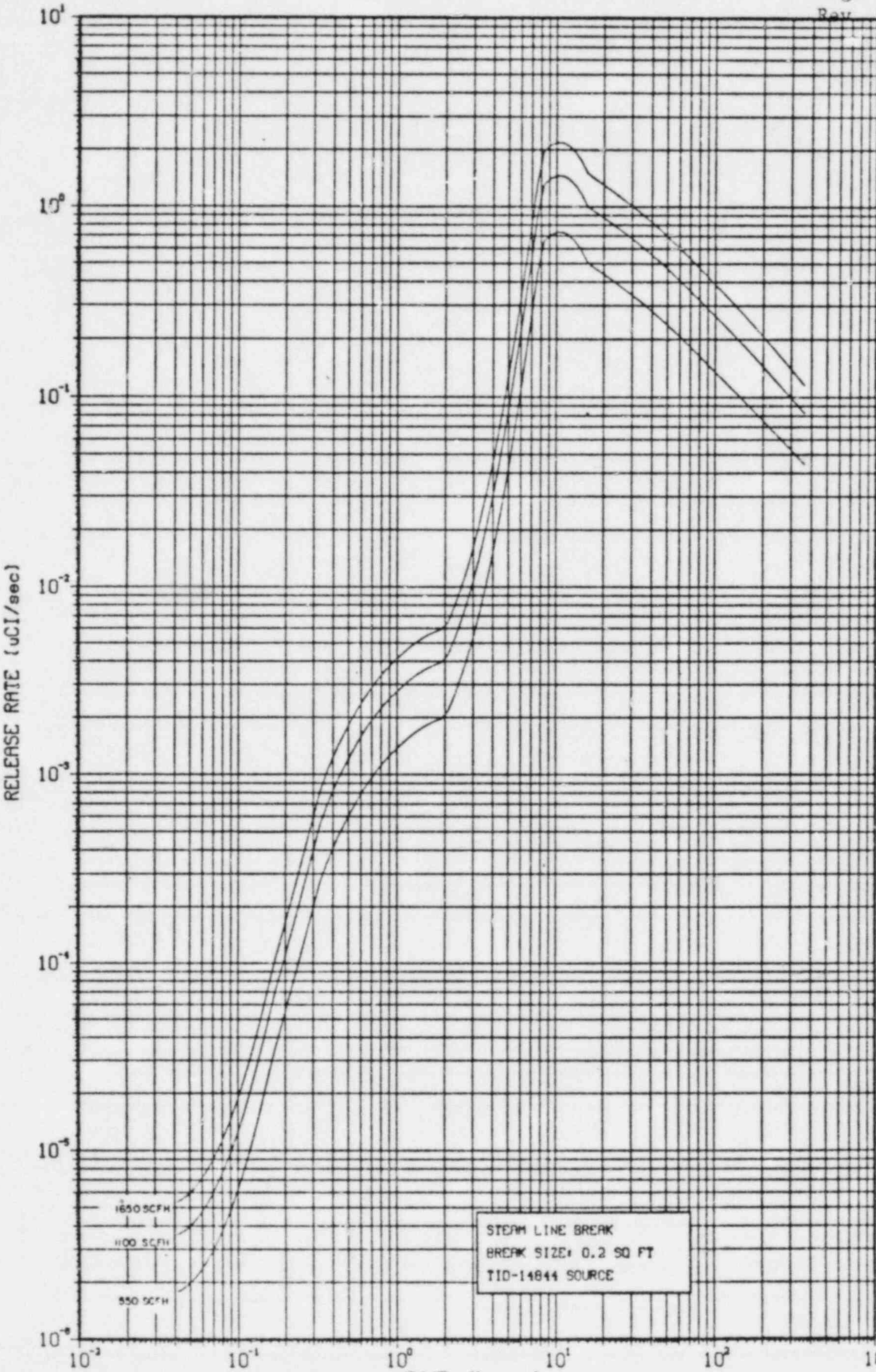


FIGURE A.5-1

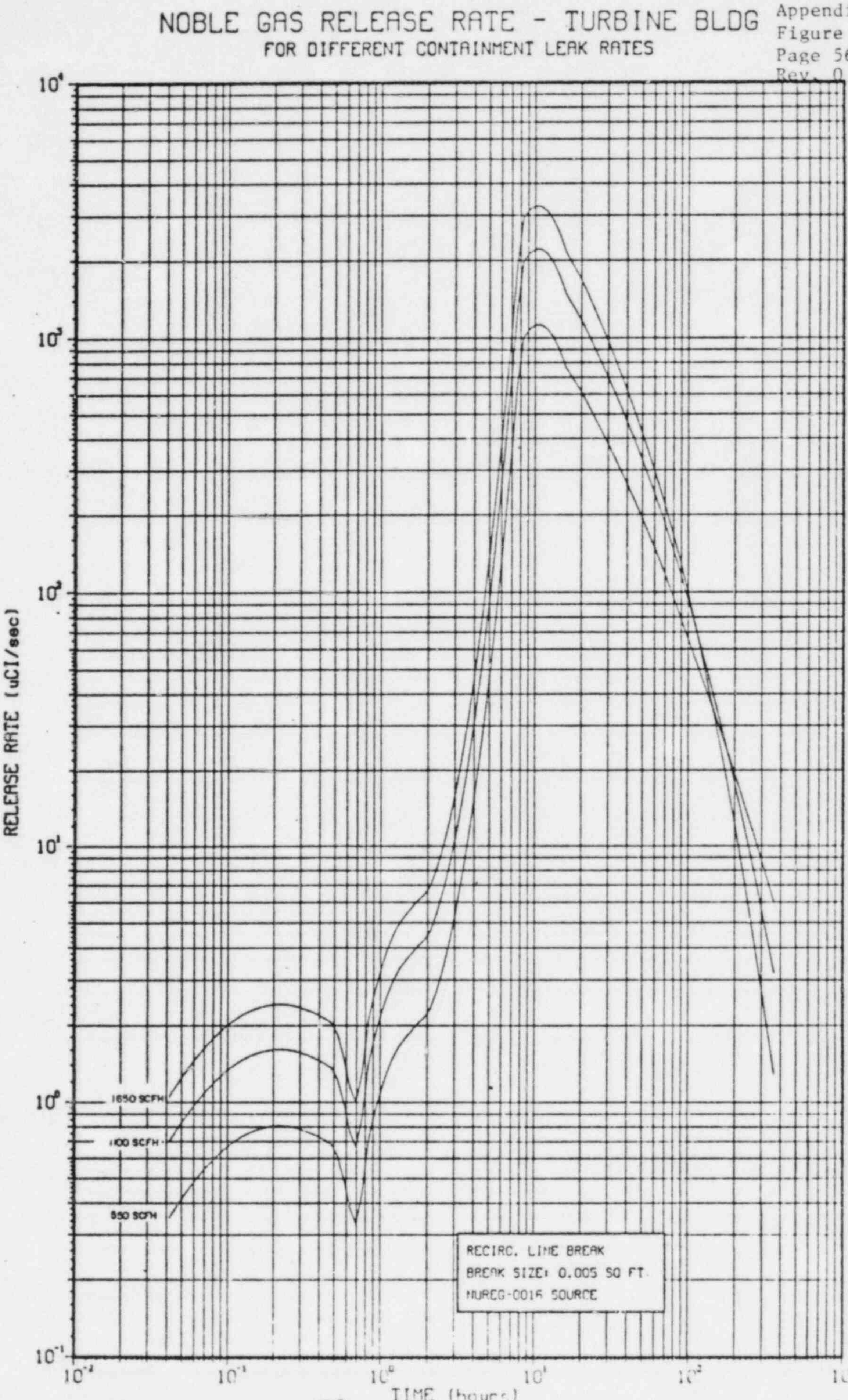


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FIGURE A.6-a

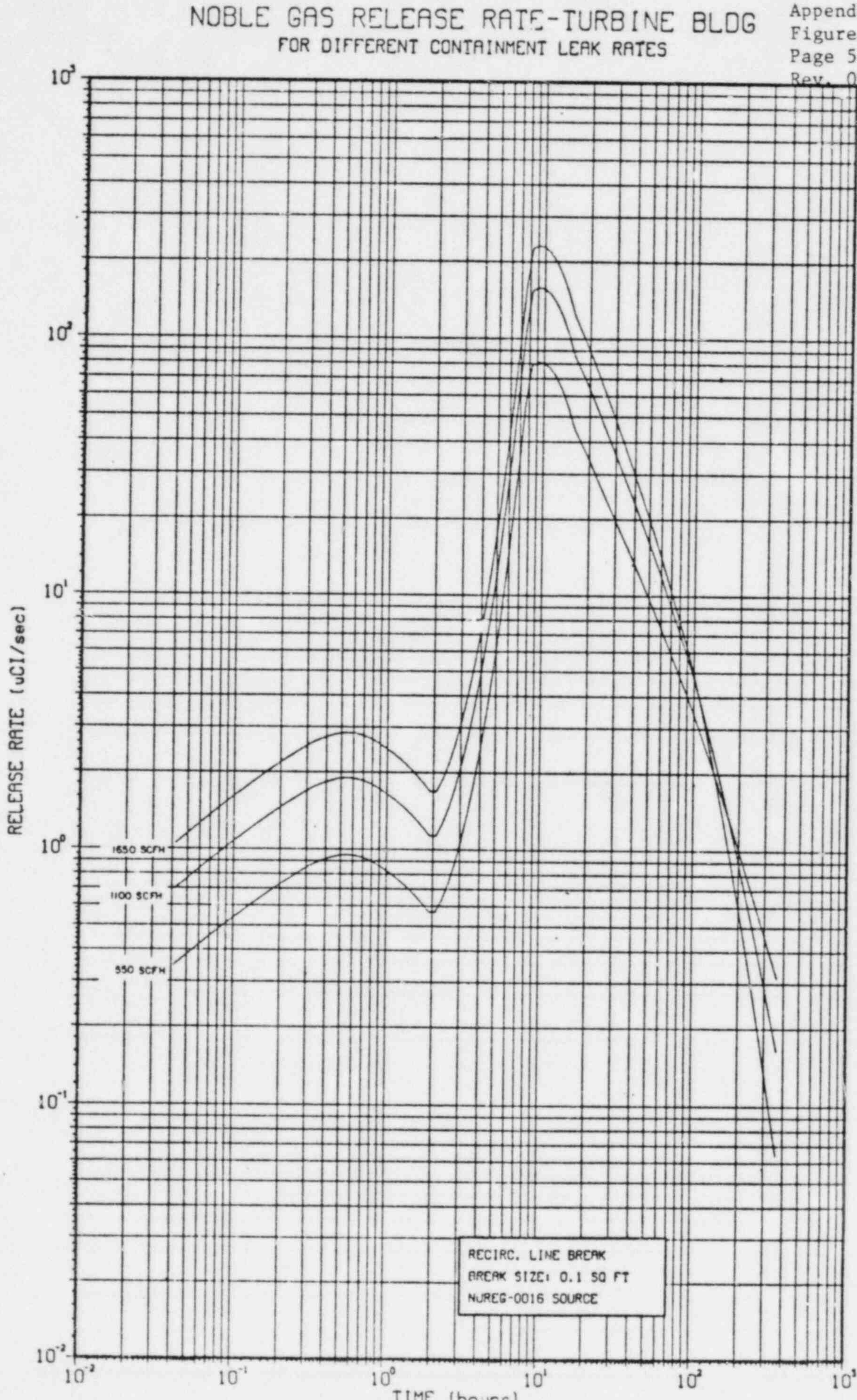
NOBLE GAS RELEASE RATE-TURBINE BLDG
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
RECIRC. LINE BREAK
BREAK SIZE: 0.1 SQ FT
NUREG-0016 SOURCE

FIGURE A.6-b

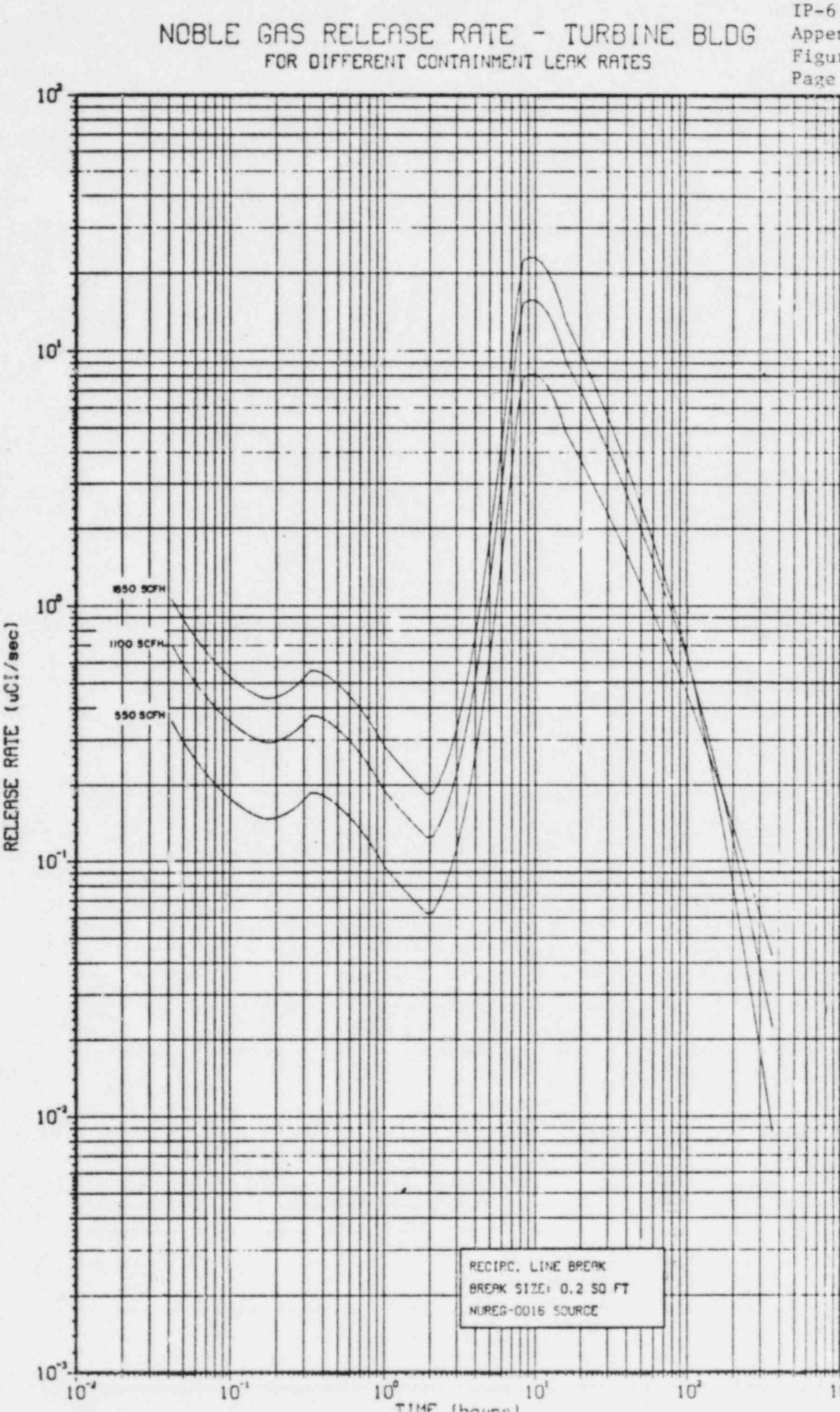
NOBLE GAS RELEASE RATE - TURBINE BLDG
FOR DIFFERENT CONTAINMENT LEAK RATES
1650 SCFH
1100 SCFH
550 SCFH
RECIRC. LINE BREAK
BREAK SIZE: 0.2 SQ FT
NUREG-0016 SOURCE
RELEASE RATE (uCi/sec)
TIME (hours)

FIGURE A6-c

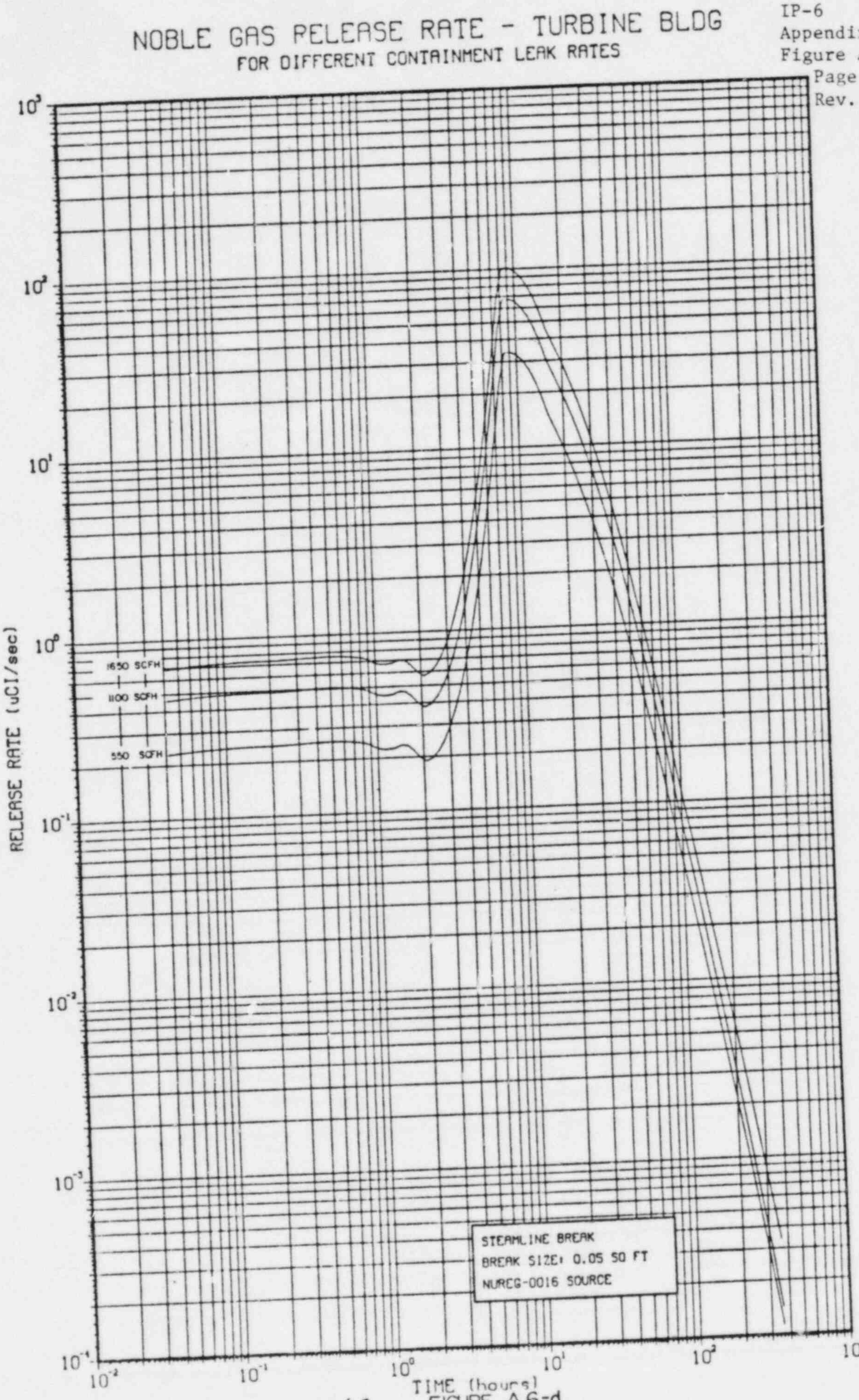
NOBLE GAS RELEASE RATE - TURBINE BLDG
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
STEAMLINE BREAK
BREAK SIZE: 0.05 SQ FT
NUREG-0016 SOURCE

FIGURE A.6-d

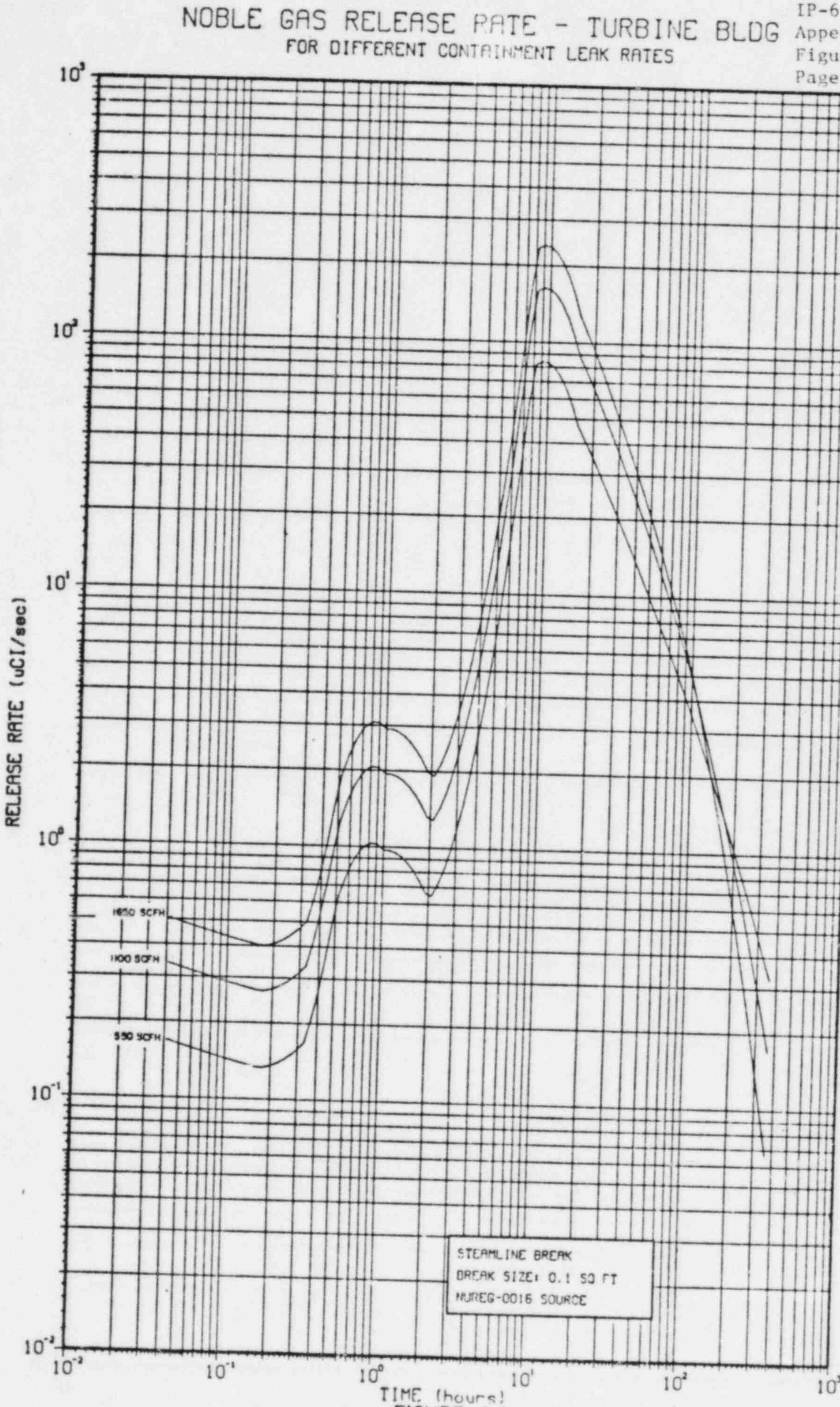
NOBLE GAS RELEASE RATE - TURBINE BLDG
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
STEAMLINE BREAK
BREAK SIZE: 0.1 SQ FT
NUREG-0016 SOURCE

FIGURE A.6-e

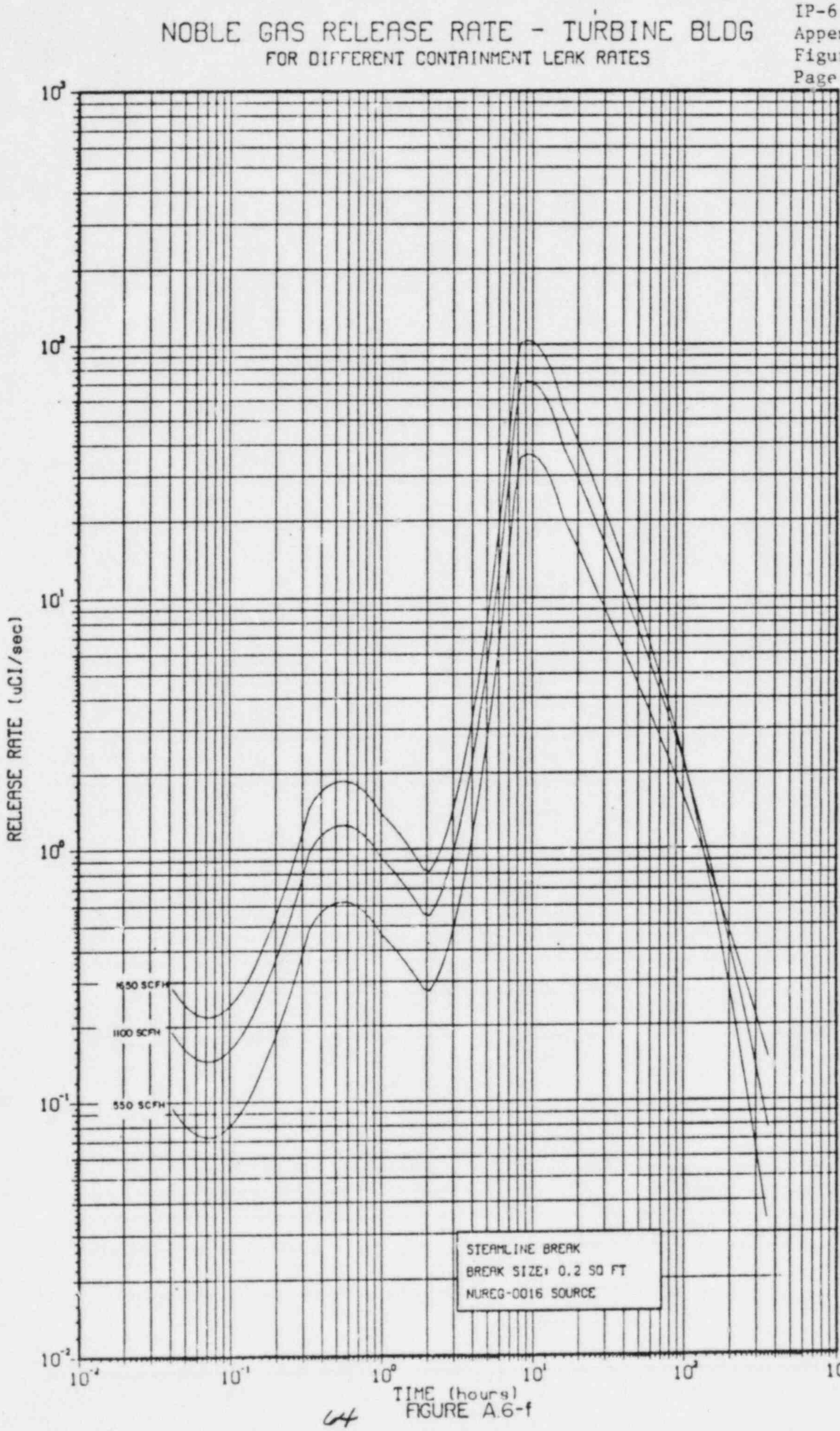
NOBLE GAS RELEASE RATE - TURBINE BLDG
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
STEAMLINE BREAK
BREAK SIZE: 0.2 SQ FT
NUREG-0016 SOURCE

FIGURE A.6-f

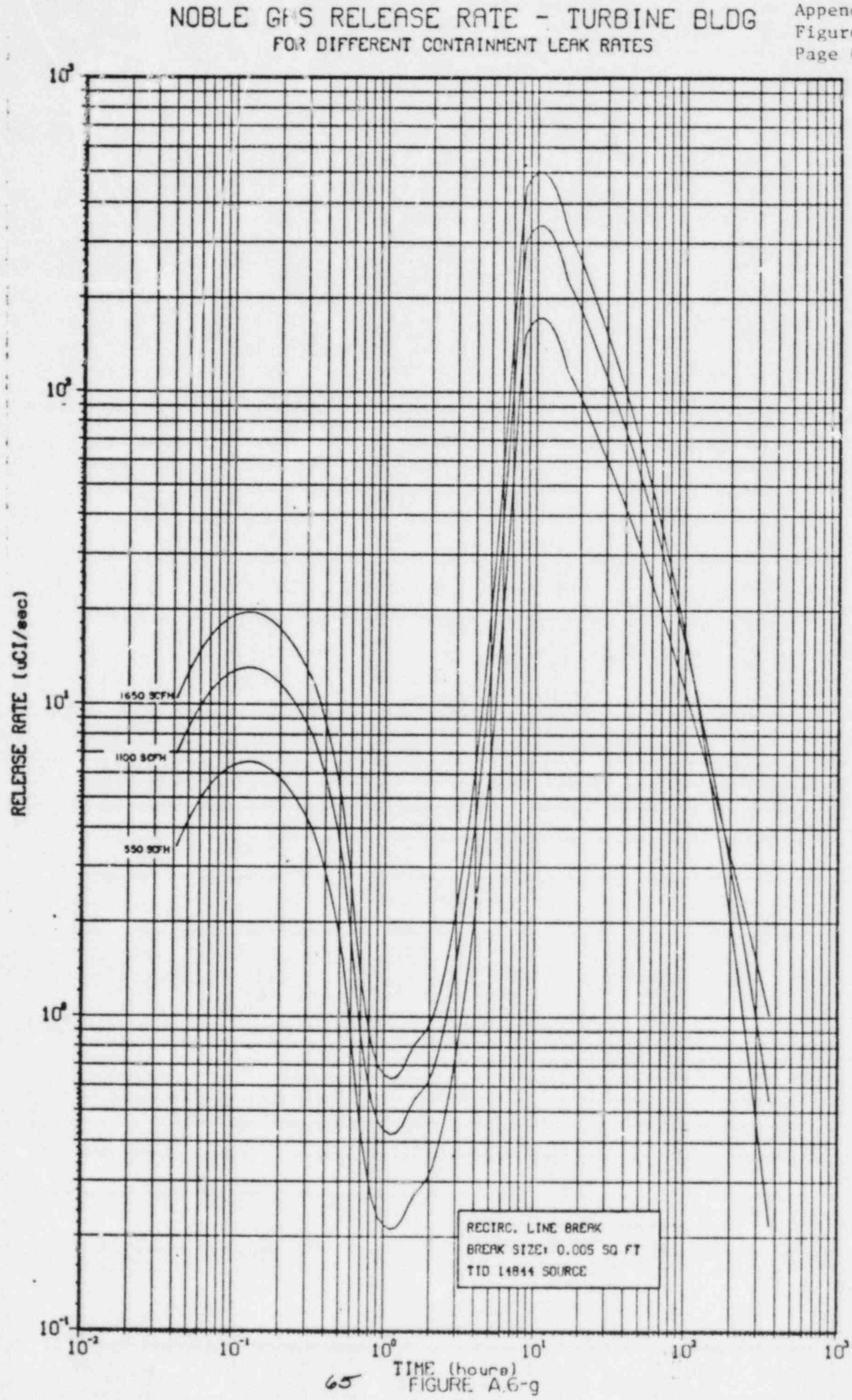

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FIGURE A.6-g

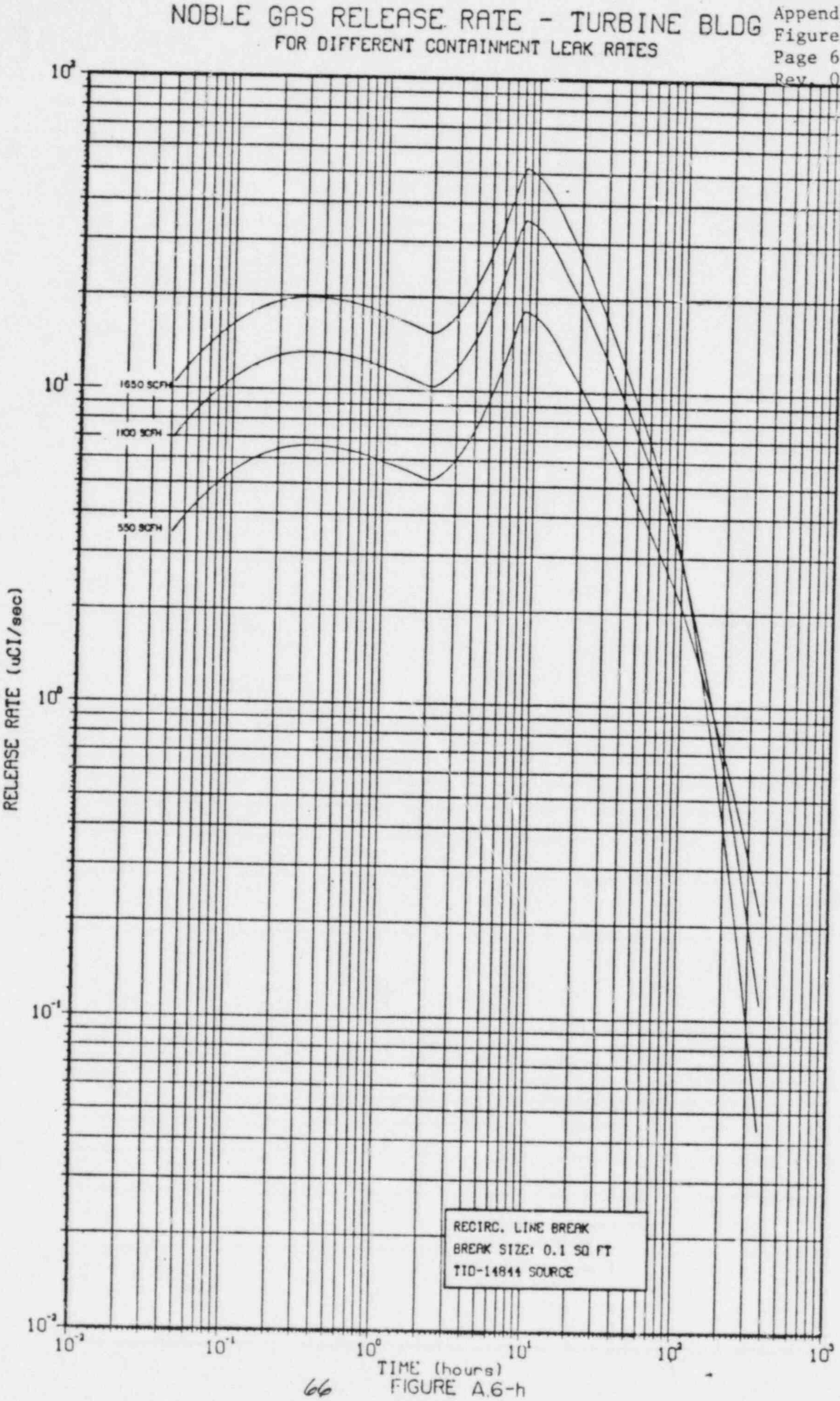

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FIGURE A.6-h

NOBLE GAS RELEASE RATE - TURBINE BLDG
FOR DIFFERENT CONTAINMENT LEAK RATES

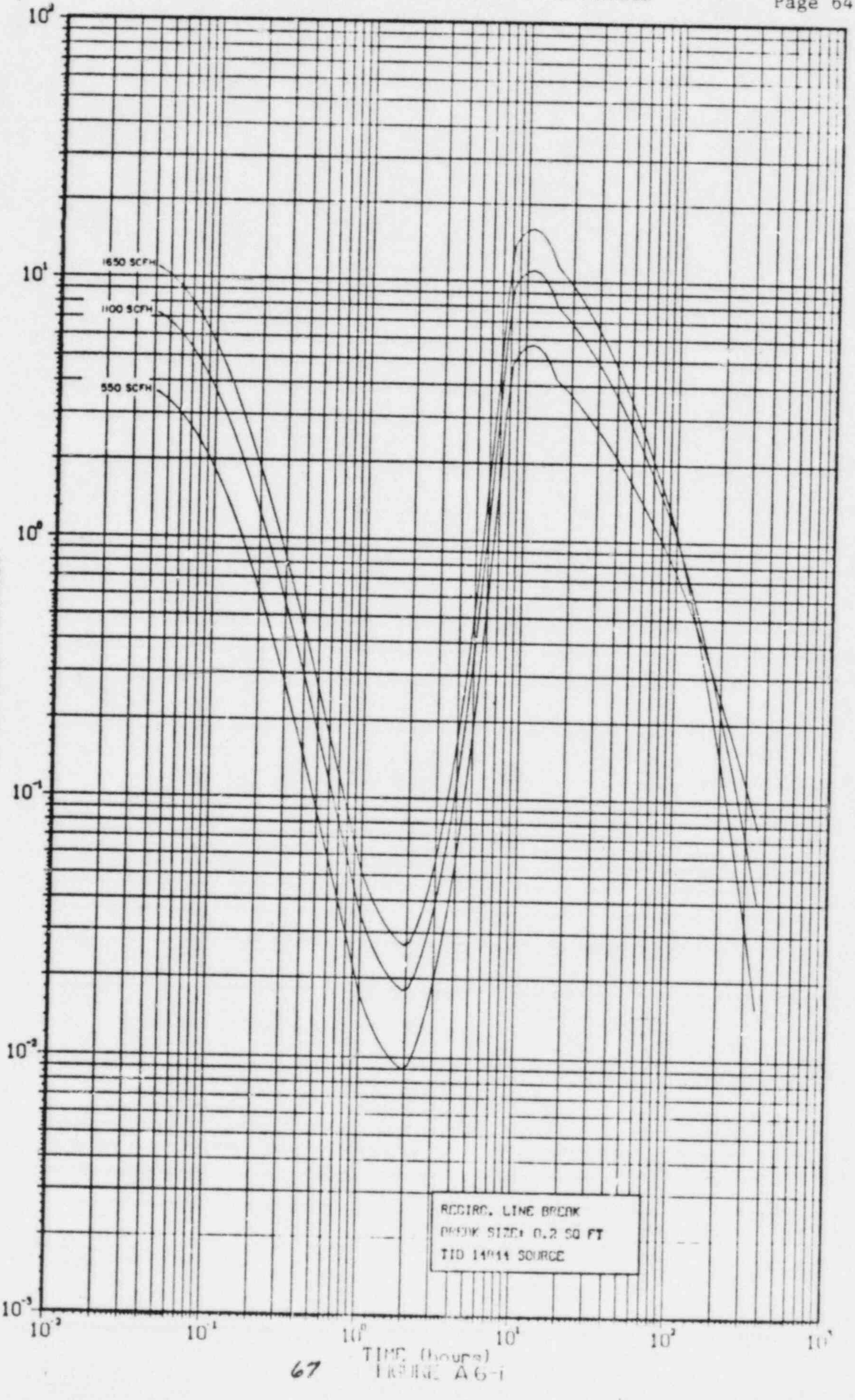

FIGURE A.6-i

NOBLE GAS RELEASE RATE - TURBINE BLDG
FOR DIFFERENT CONTAINMENT LEAK RATES

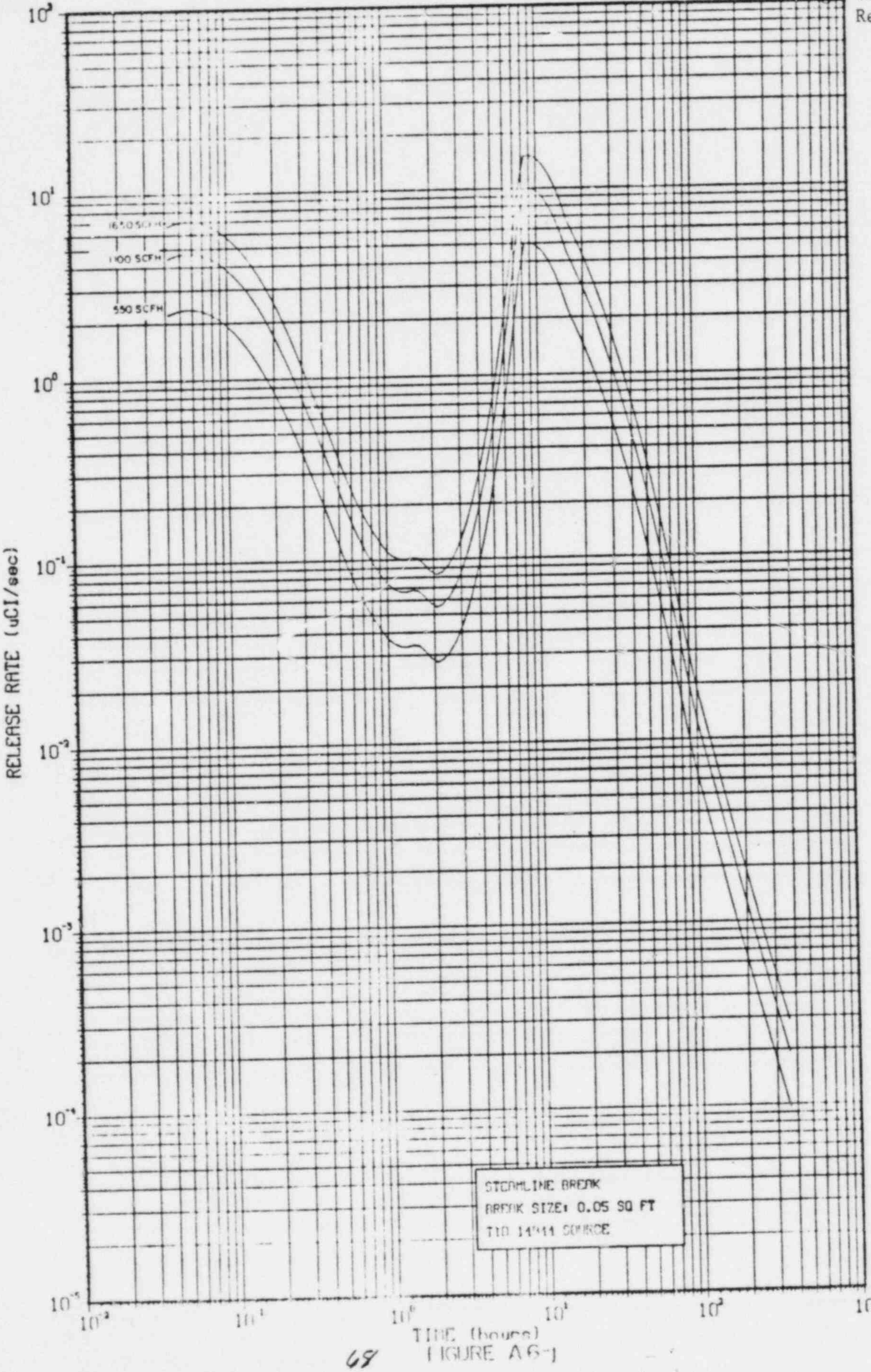

FIGURE A6-j

NOBLE GAS RELEASE RATE - TURBINE BLDG

FOR DIFFERENT CONTAINMENT LEAK RATES

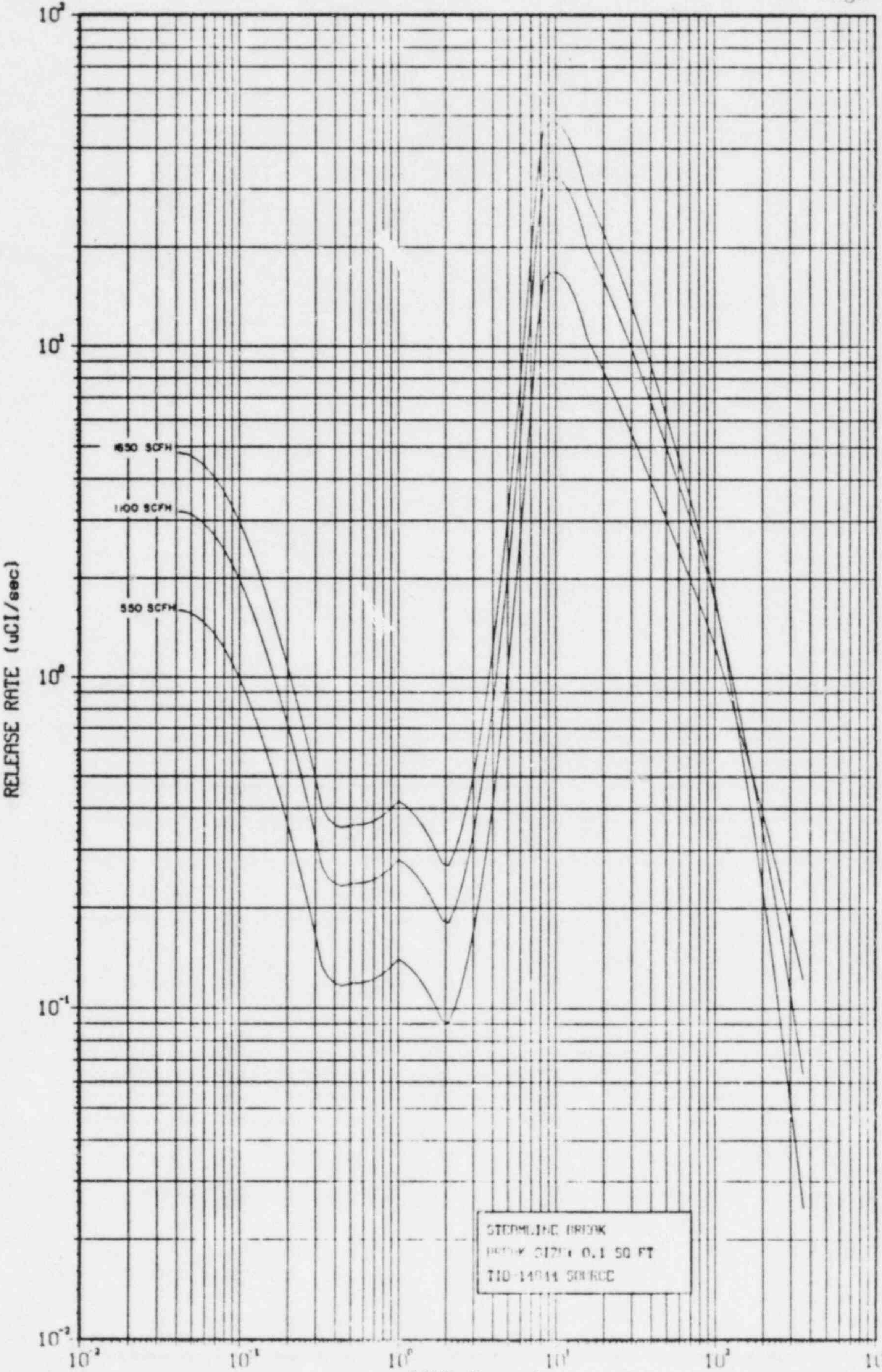

FIGURE A.6-k

NOBLE GAS RELEASE RATE - TURBINE BLDG

FOR DIFFERENT CONTAINMENT LEAK RATES

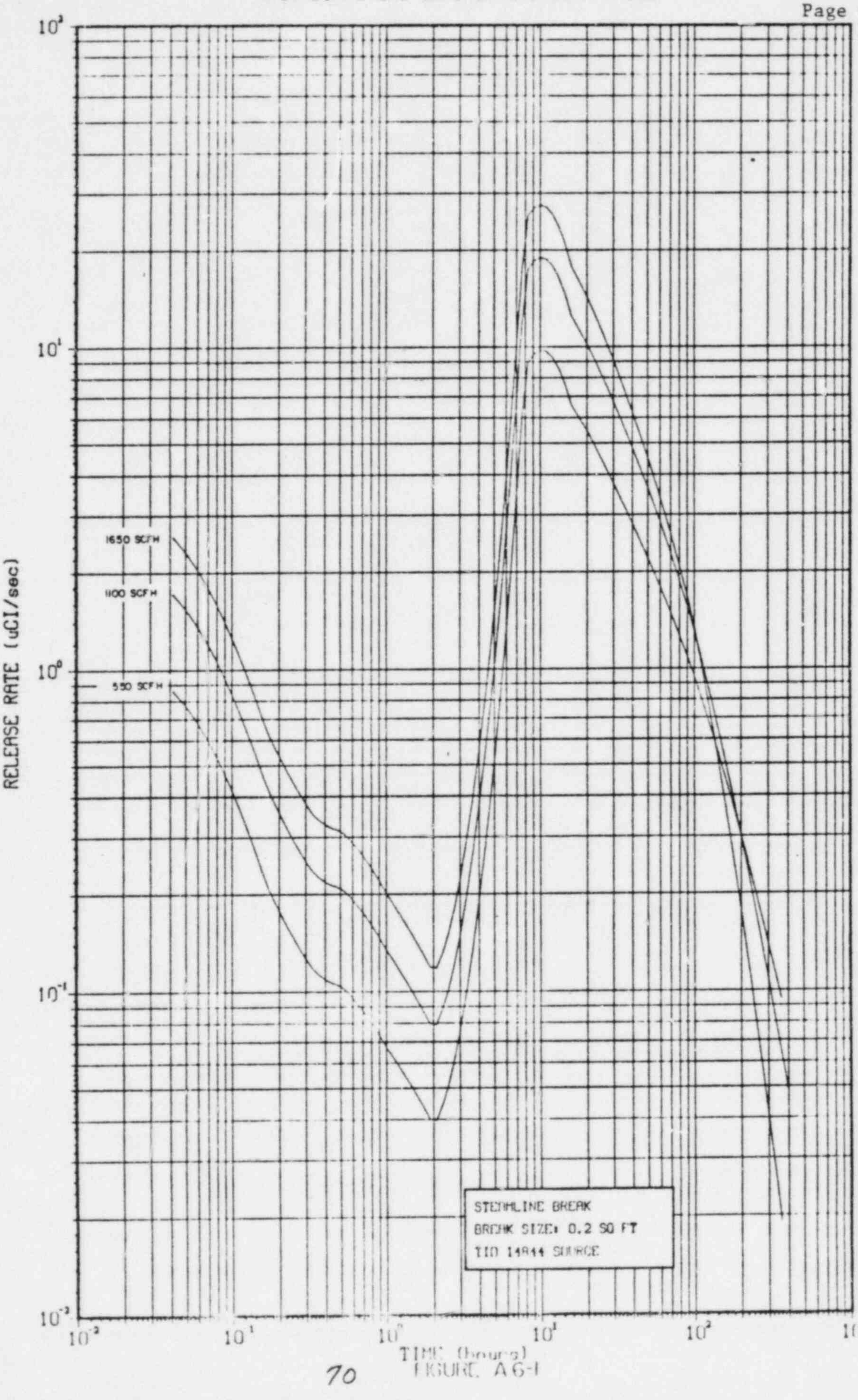

FIGURE A6-1

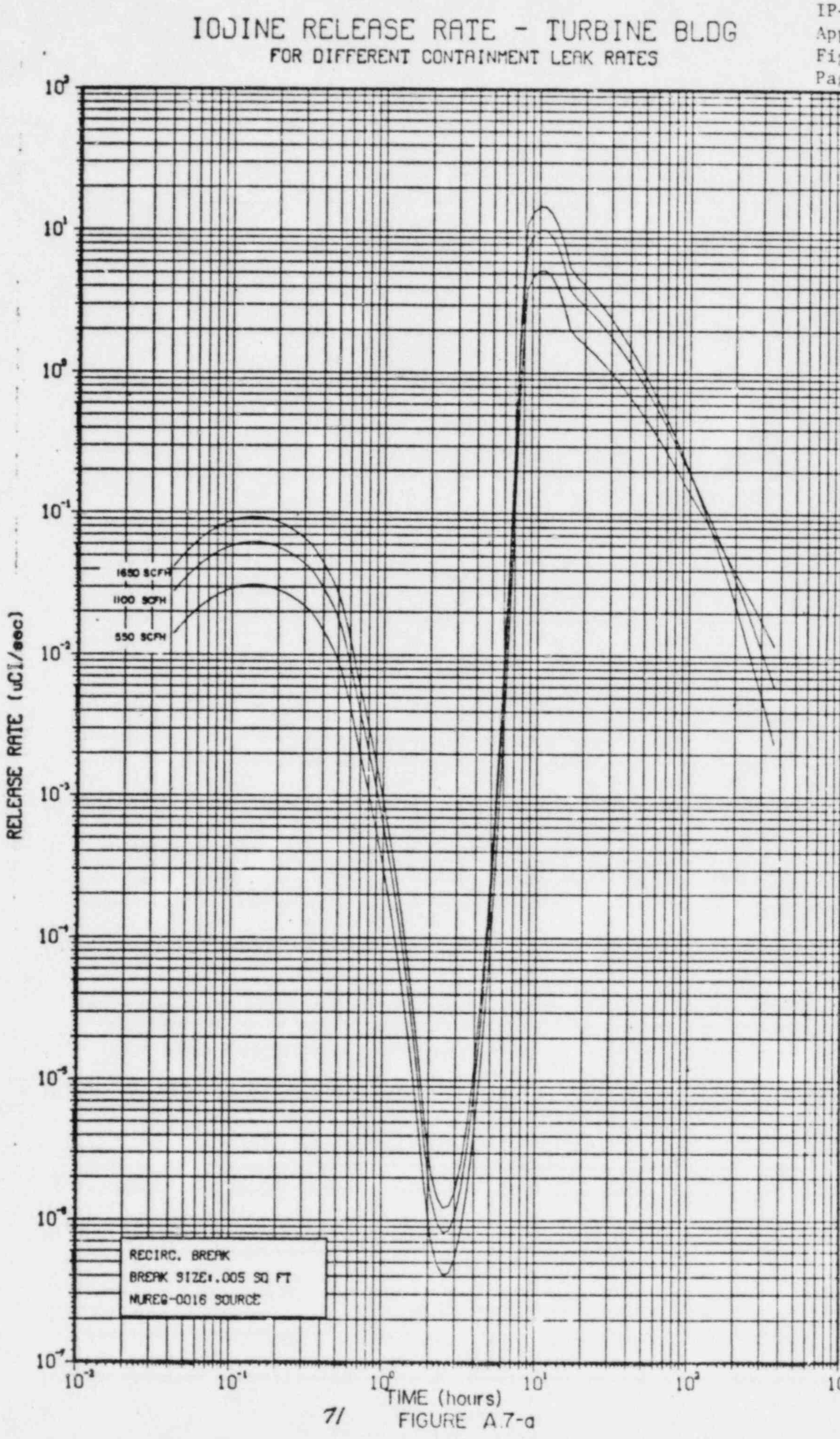
IODINE RELEASE RATE - TURBINE BLDG
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCi/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
RECIRC. BREAK
BREAK SIZE: .005 SQ FT
NUREG-0016 SOURCE

FIGURE A.7-a

IODINE RELEASE RATE - TURBINE BLDG

FOR DIFFERENT CONTAINMENT LEAK RATES

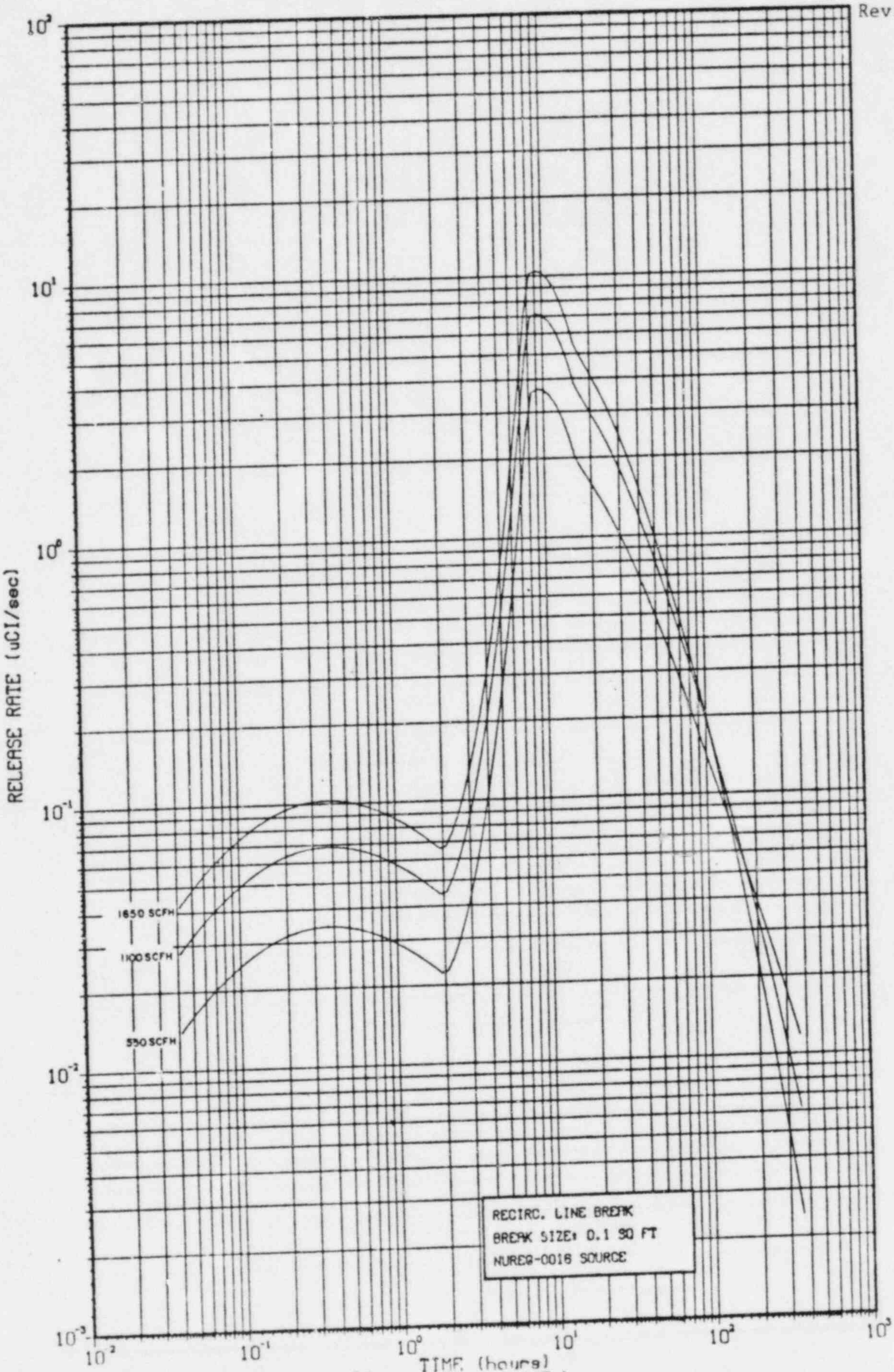

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FIGURE A.7-b

IODINE RELEASE RATE - TURBINE BLDG

FOR DIFFERENT CONTAINMENT LEAK RATES

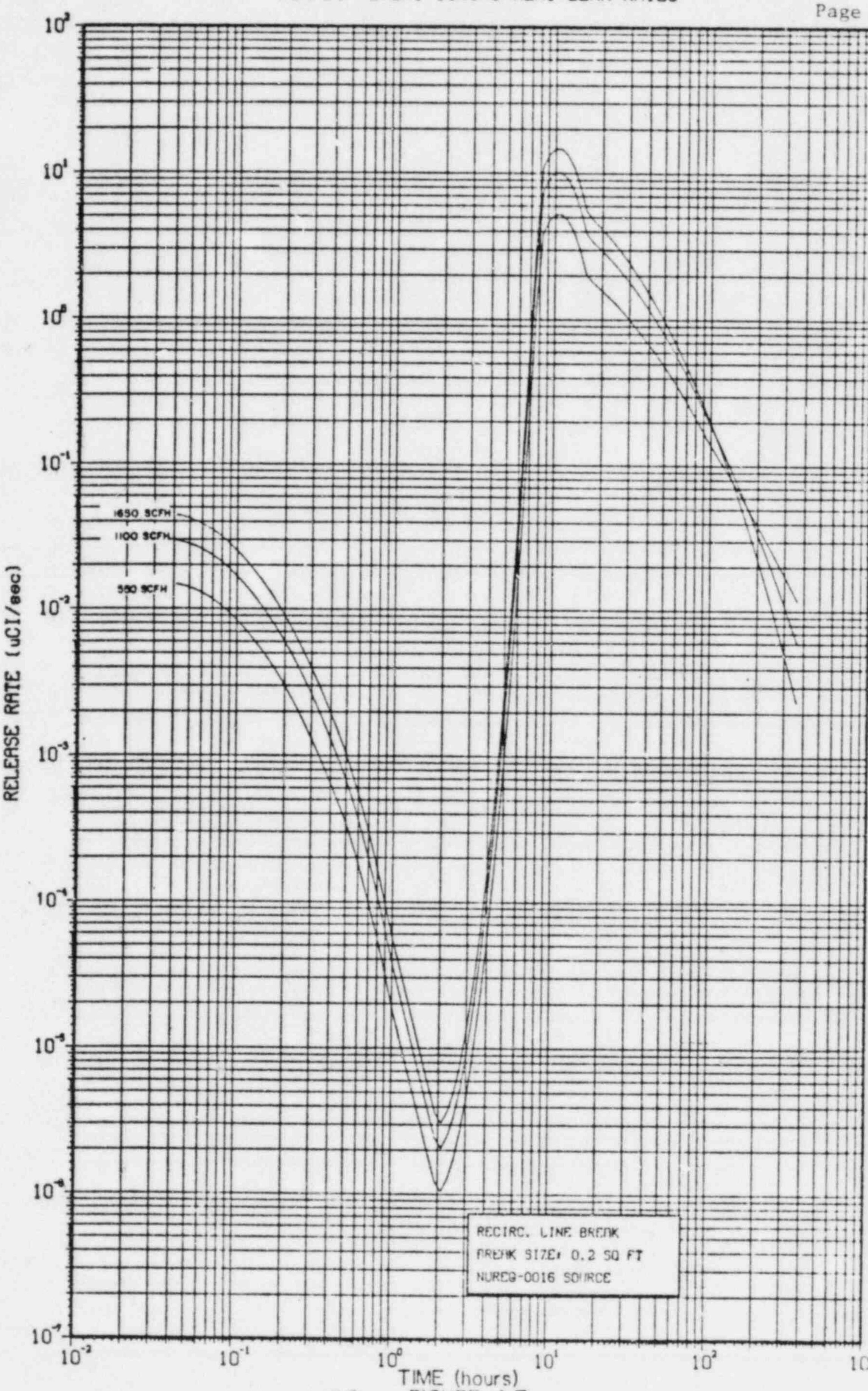

FIGURE A.7-c

IODINE RELEASE RATE - TURBINE BLDG

FOR DIFFERENT CONTAINMENT LEAK RATES

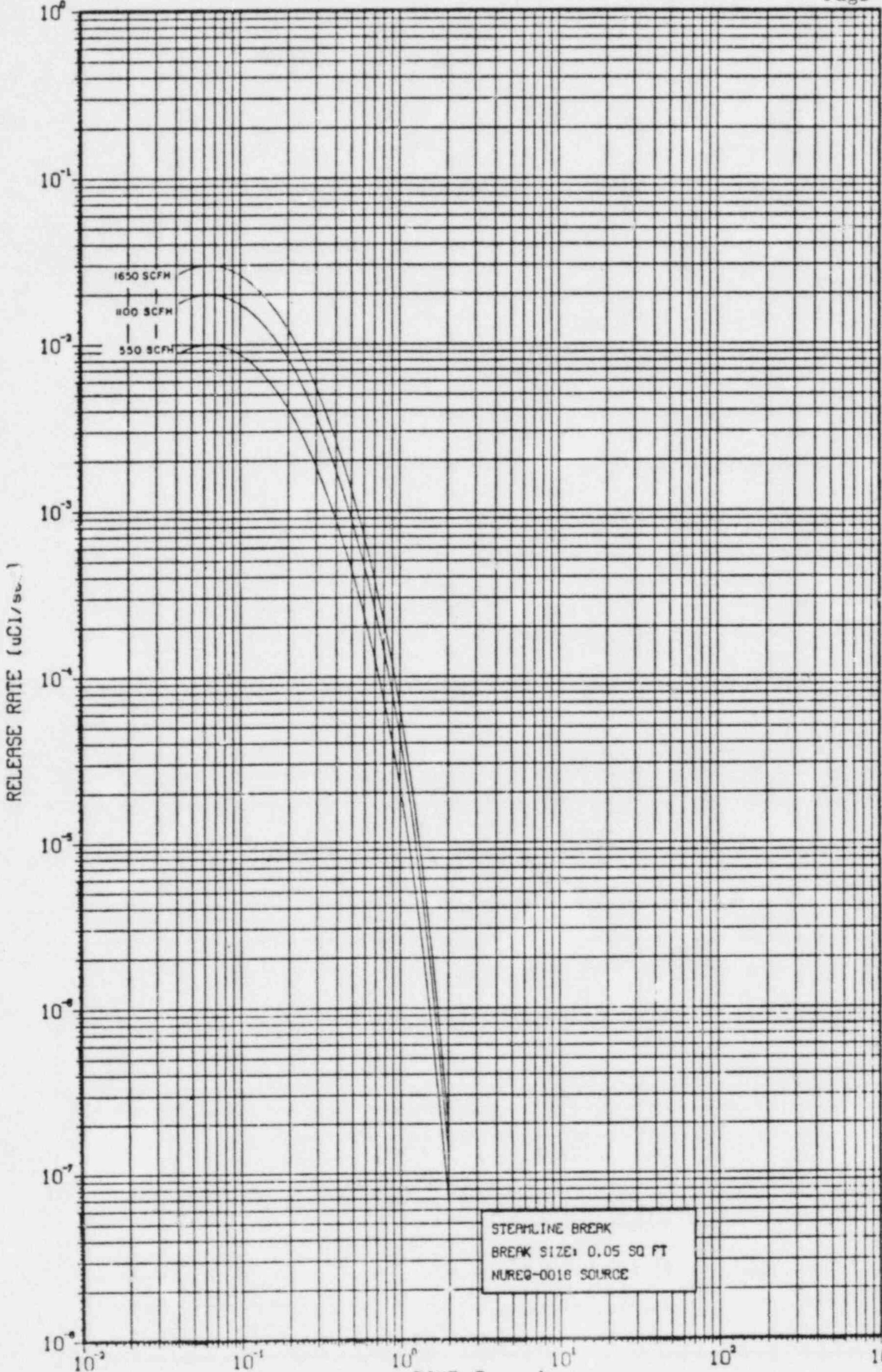

FIGURE A.7-d

IODINE RELEASE RATE - TURBINE BLDG

FOR DIFFERENT CONTAINMENT LEAK RATES

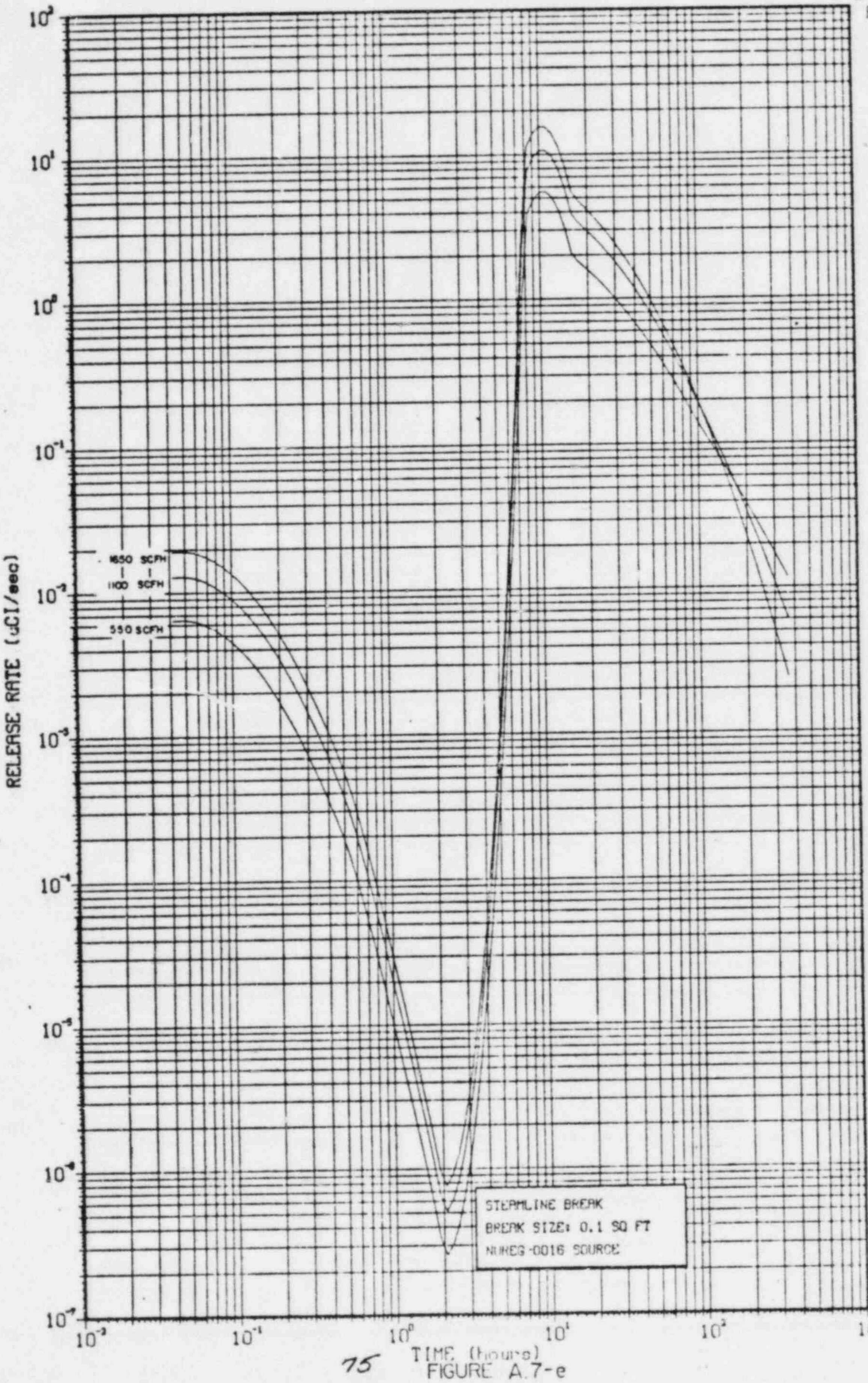

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FIGURE A.7-e

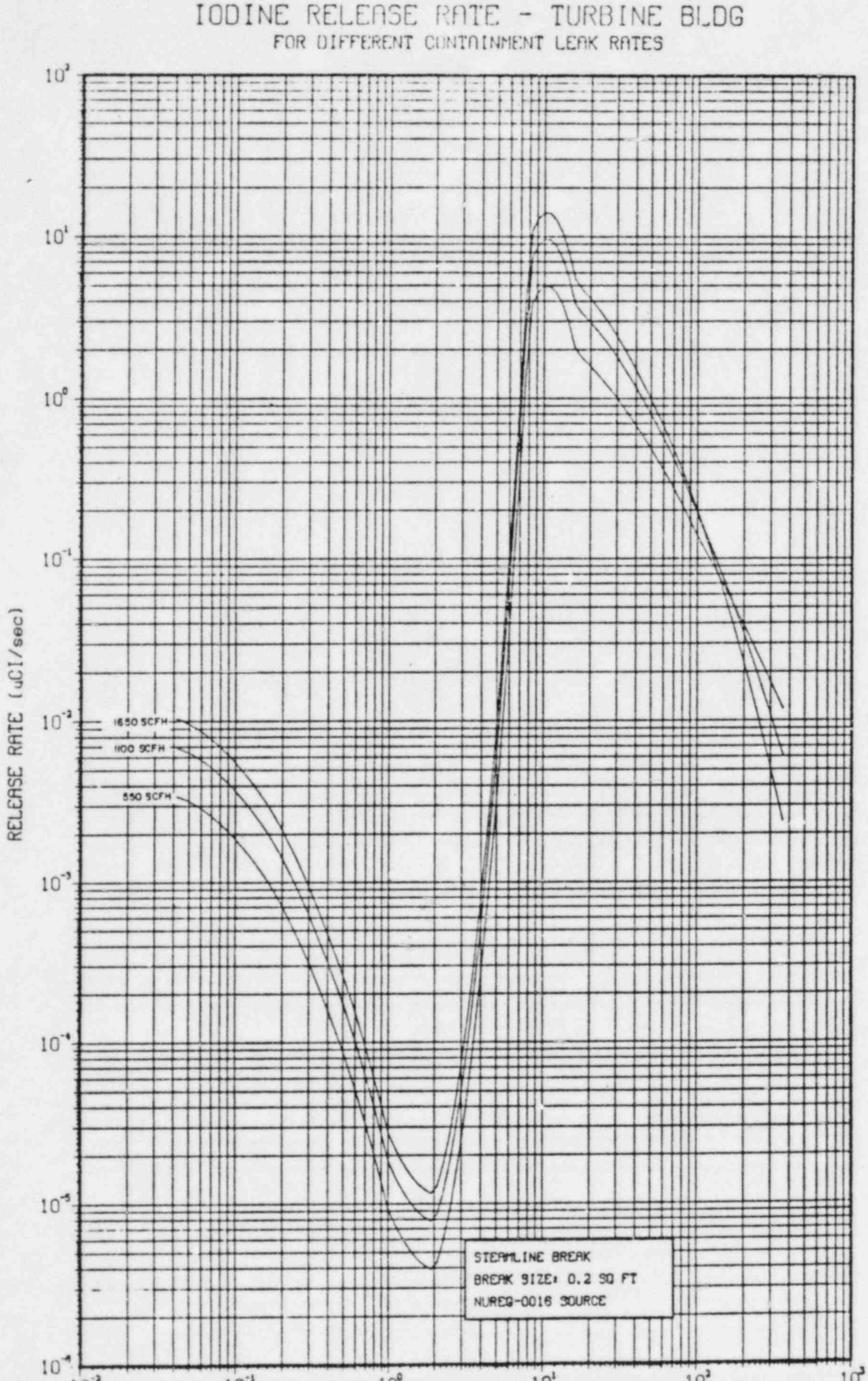

FIGURE A.7-f

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IODINE RELEASE RATE - TURBINE BLDG
FOR DIFFERENT CONTAINMENT LEAK RATES

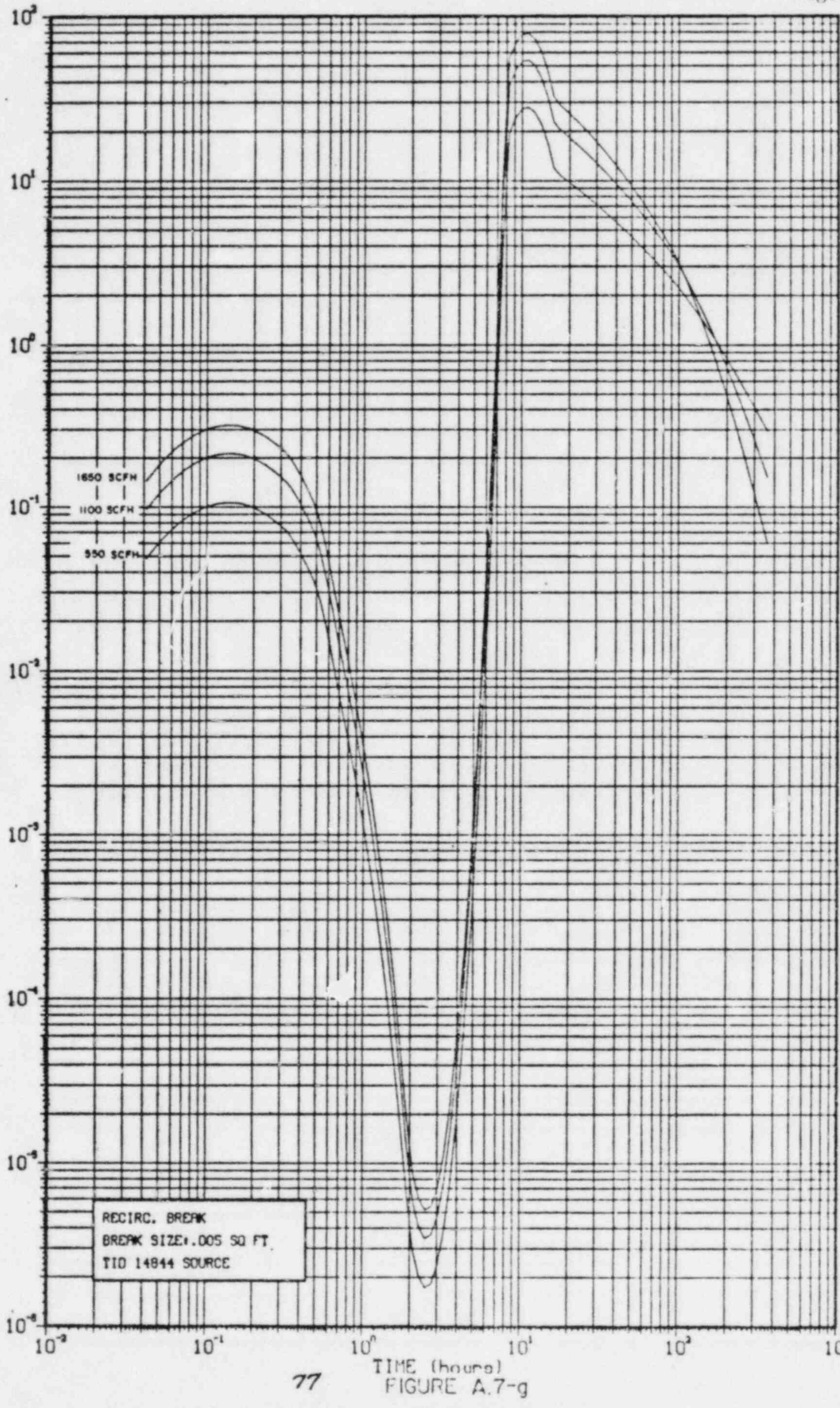


FIGURE A.7-g

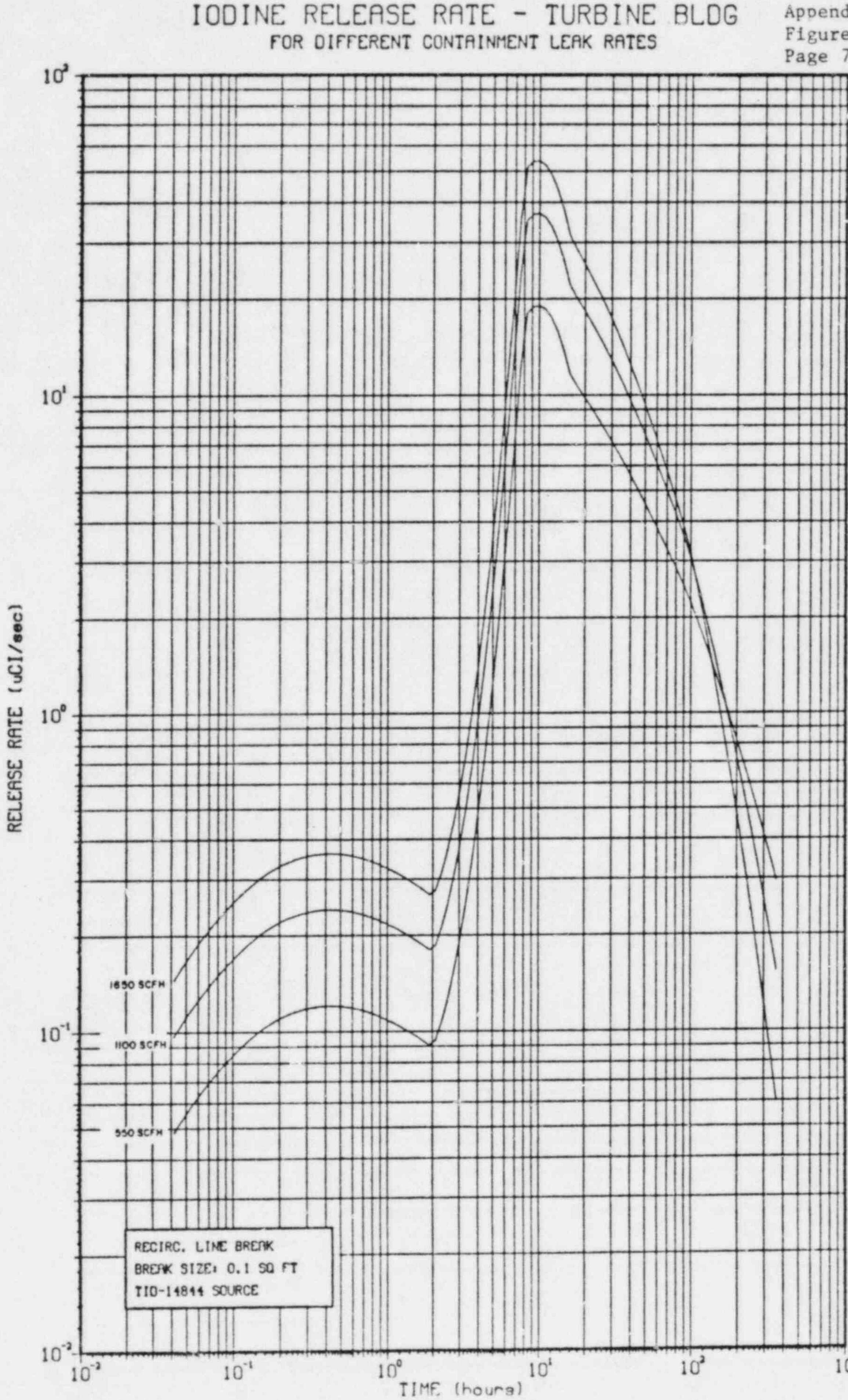
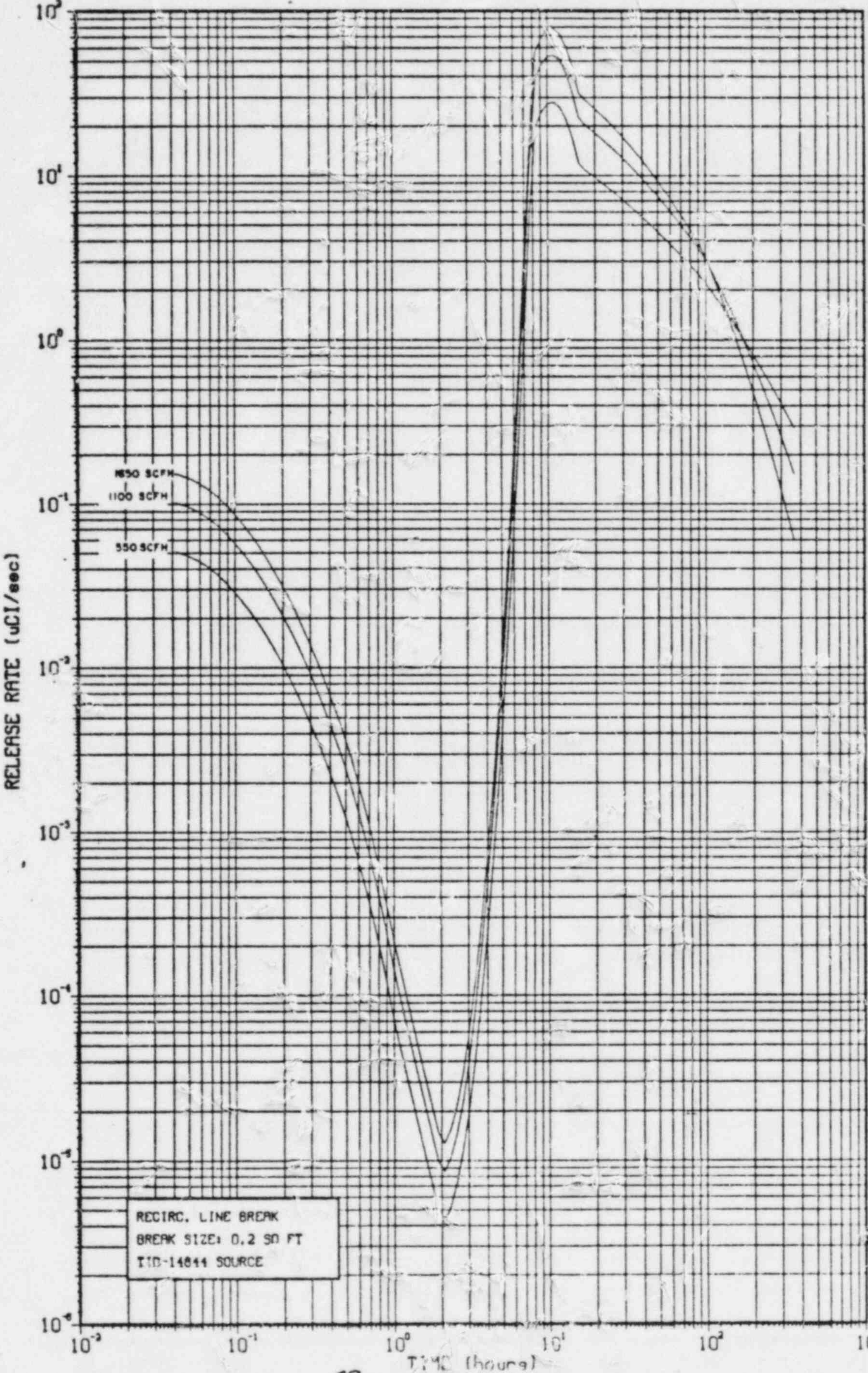

IODINE RELEASE RATE - TURBINE BLDG
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCi/sec)
TIME (hours)
1650 SCFH
1100 SCFH
550 SCFH
RECIRC. LINE BREAK
BREAK SIZE: 0.1 SQ FT
TID-14844 SOURCE

FIGURE A.7-h

IODINE RELEASE RATE - TURBINE BLDG
FOR DIFFERENT CONTAINMENT LEAK RATES



79

FIGURE A7-1

IODINE RELEASE RATE - TURBINE BLDG

FOR DIFFERENT CONTAINMENT LEAK RATES

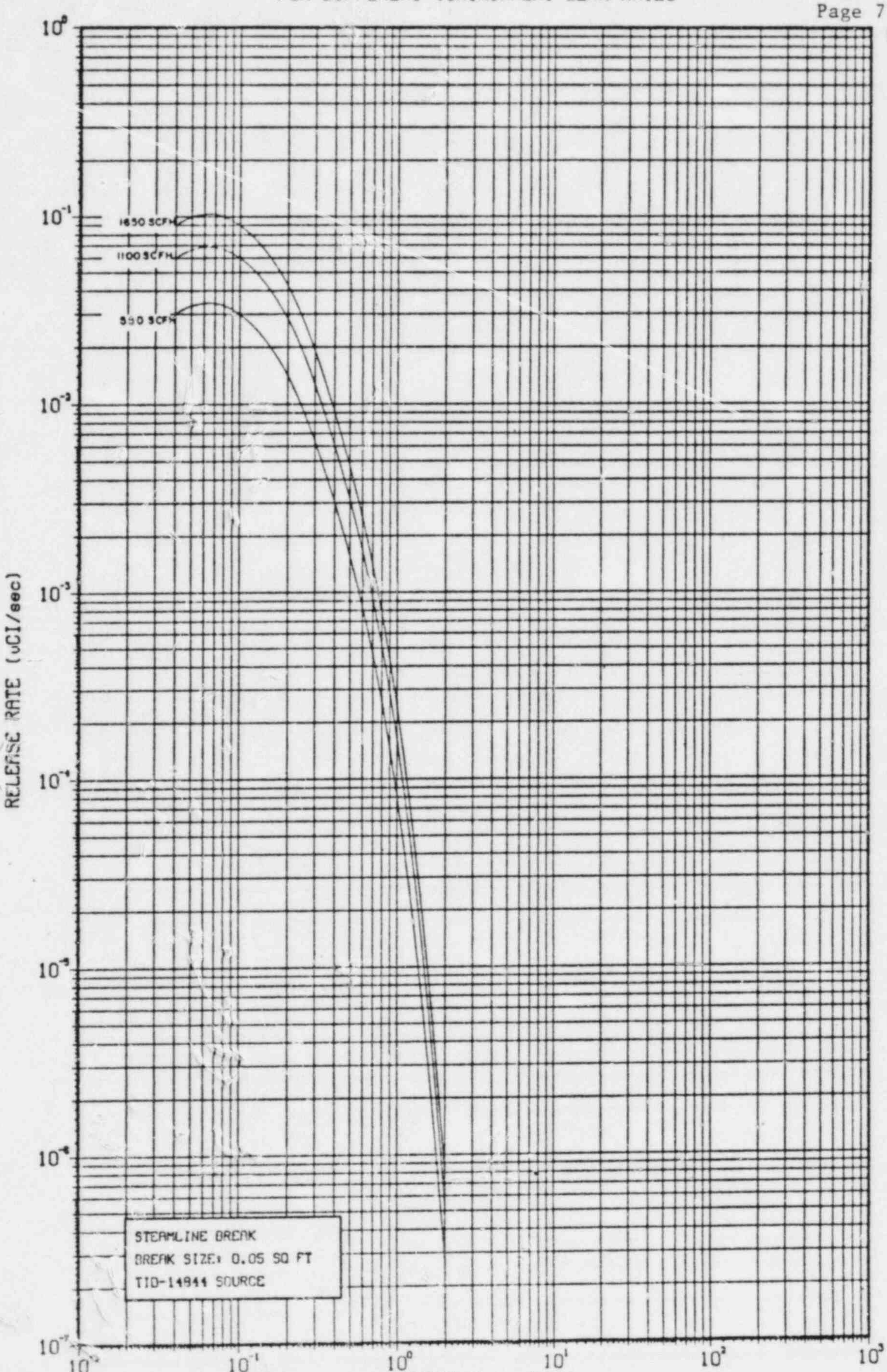


FIGURE A.7-j

IODINE RELEASE RATE - TURBINE BLDG

FOR DIFFERENT CONTAINMENT LEAK RATES

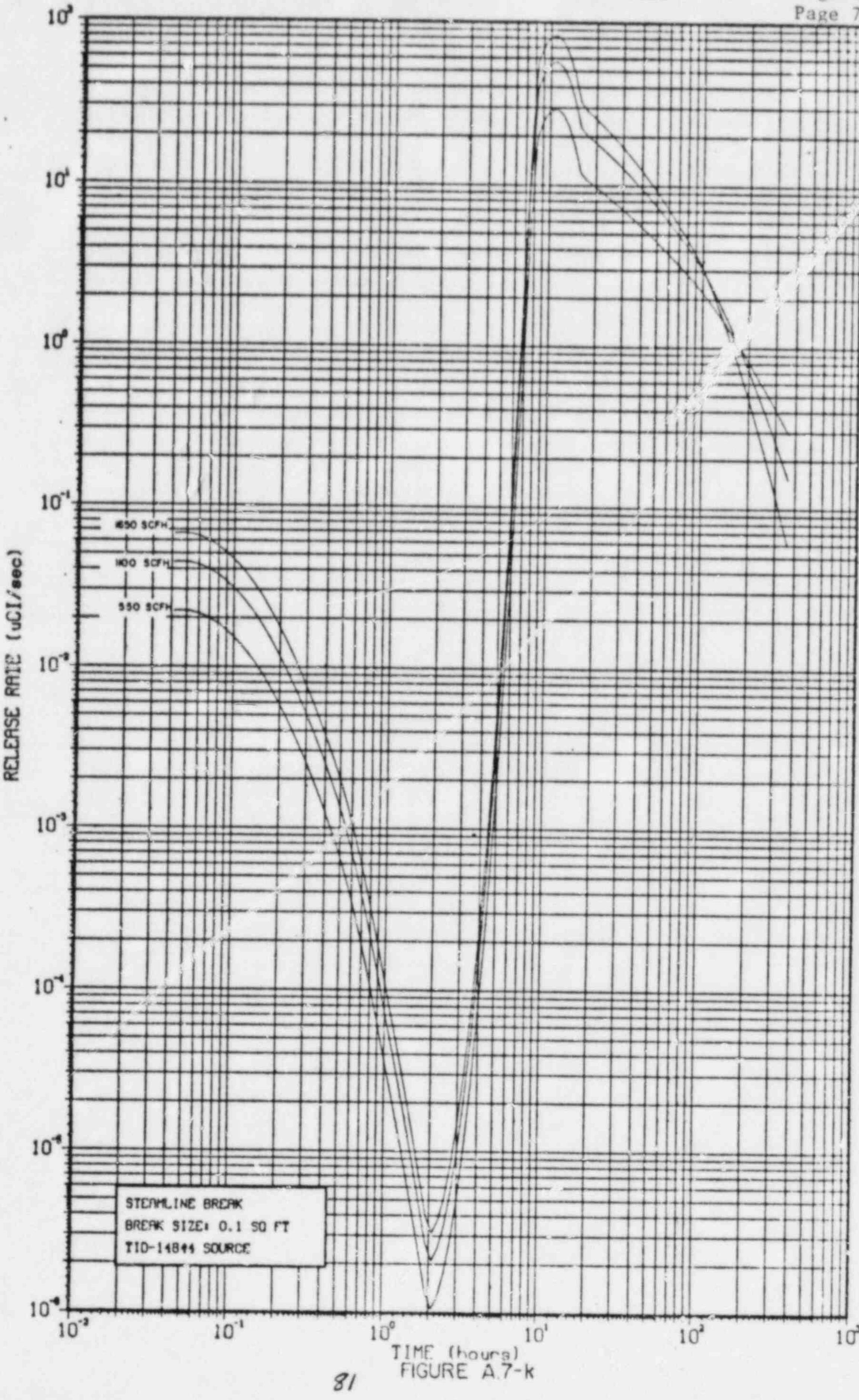


FIGURE A.7-k

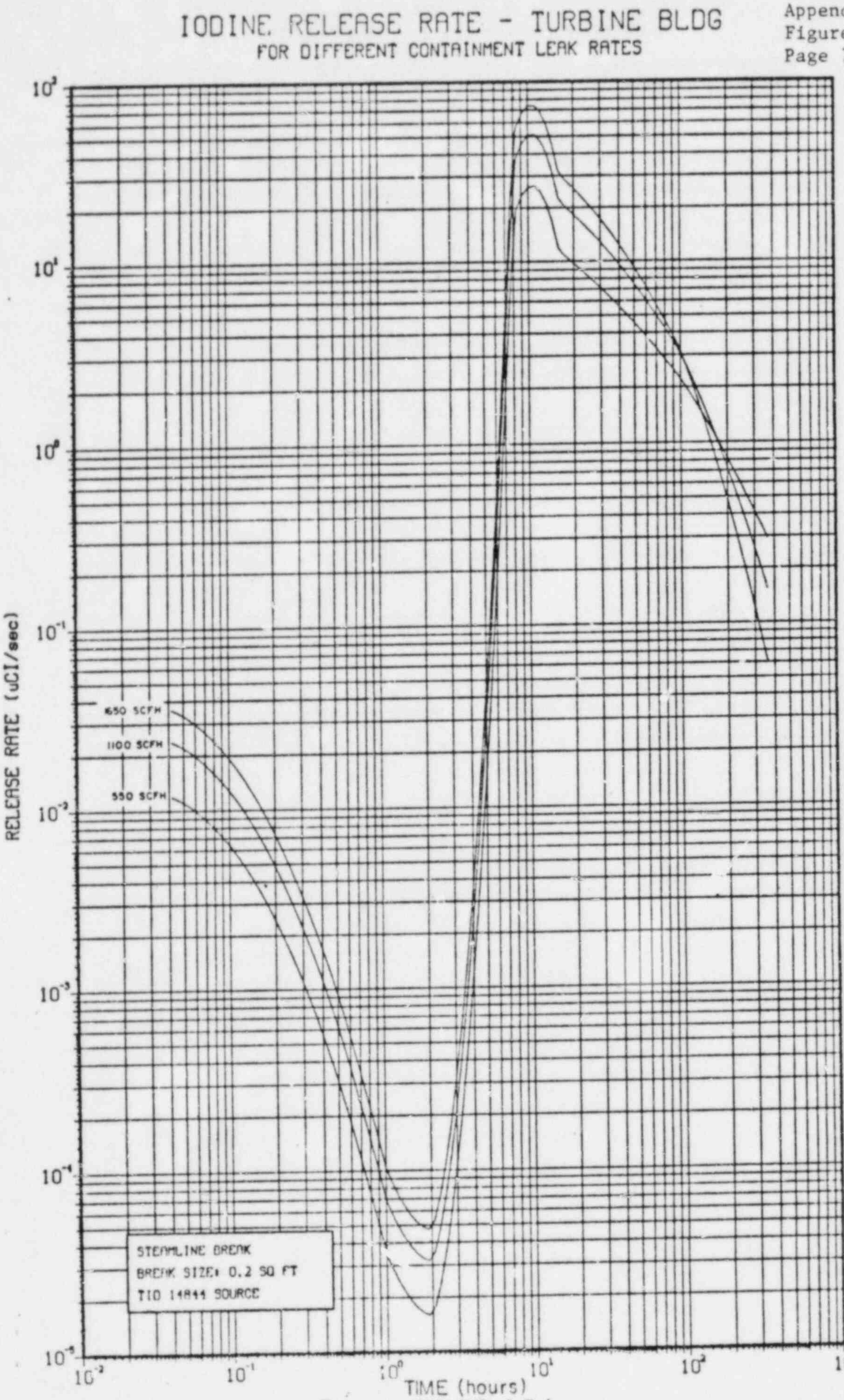

IODINE RELEASE RATE - TURBINE BLDG
FOR DIFFERENT CONTAINMENT LEAK RATES
RELEASE RATE (uCI/sec)
TIME (hours)
650 SCFH
1100 SCFH
550 SCFH
STEAMLINE BREAK
BREAK SIZE: 0.2 SQ FT
TID 14844 SOURCE
10^{2}
10^{1}
10^{0}
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-2}
10^{-1}
10^{0}
10^{1}
10^{2}
10^{3}

FIGURE A.7-1

TVA Predictive Model
Appendix B - SQN

1.0 PLANT PARAMETERS REQUIRED FOR RELEASE RATE CALCULATION

1.1 Determine which isotope spectrum to use based on the latest primary coolant measurement.

1.1.1 On Worksheet 1-A record the latest measured primary coolant activity (per SI-54 or postaccident measurement) and calculate the activity ratios.

1.1.2 On Table 1.1, circle the entries corresponding to the ratios calculated in 1.1.1.

1.1.3 On Table 1.1, circle the isotope spectrum on the line with the most circled entries. This isotope spectrum will be used in all steps of 1.0 and 2.0 of this procedure.

IMPORTANT: A POWER CHANGE OF MORE THAN 15 PERCENT WITHIN 2-3 HOURS BEFORE THE PRIMARY COOLANT SAMPLE WAS TAKEN INVALIDATES THE RATIOS INVOLVING IODINE (I). IN SUCH A CASE, RELY ON THE C/R RATIO.

1.2 Determine Normalization Factor

The information labeled "Expected" and "Design" is based on 1 percent failed fuel; the "Modified TID-14844" information is based on 1 percent of the total radioactivity inventory of an equilibrium reactor core.

1.2.1 On Worksheet 1-A, circle the calculated specific activities for each of the isotopes (I-131, Cs-137, Rb-88) given in Table 1.2 for the isotope spectrum chosen in 1.1.3.

1.2.2 Calculate and record the ratio of the measured specific activity (from step 1.1.1) to the circled calculated values of Table 1.2.

1.2.3 Select a normalization factor corresponding most closely to two of the three ratios. Record this on worksheet 1-A, step 1.2.3.

Note: For "Expected" and "Design" isotope spectra, the normalization factor corresponds to the failed fuel fraction; for the "Modified TID-14844" spectrum, the normalization factor corresponds to the percentage of equilibrium core inventory.

1.3 Determine primary coolant leak rate into containment.

Use worksheets 1-B and 1-C. Record readings of radiation monitors RM-90-100, RM-90-112, RM-90-106. If RM-90-112 and/or RM-90-106 readings are off-scale, record RM-90-2. Record the time the readings are taken. It should be noted that all radiation monitor readings are assumed to be background subtracted before being used in this predictive model. Readings taken immediately prior to the accident should be used as the background values if available. This applies to effluent as well as area monitors.

1.3.1 From the last detector efficiency verification (TI-18, Worksheet 18-C-1B), convert cpm readings to Ci/cc.

1.3.2 Calculate the elapsed time from reactor shutdown to the reading of monitor readings.

1.3.3 If both radiation monitors RM-90-112 and RM-90-106 readings are off-scale, skip to 1.3.7; otherwise, continue with 1.3.4.

1.3.4 From the latest detector efficiency verfication (TI-18, Worksheet 18-C-1B) calculate the ratio of Xe-133 cpm to total cpm for RM-90-112, RM-90-106.

1.3.5.a Calculate the Xe-133 specific activity in upper containment by multiplying radiation monitor RM-90-112 reading by its Xe-133/total ratio.

1.3.5.b Calculate the Xe-133 specific activity in lower containment by multiplying radiation monitor RM-90-106 reading by its Xe-133/total ratio.

1.3.6 Determine the primary coolant leak rate into lower containment as follows:

a. For isotopic spectrum:

Expected - Use Figure B.1-a.

Design - Use Figure B.1-b.

Modified TID-14844 - Use Figure B.1-c.

b. Normalize the measured Xe-133 specific activity by dividing by the normalization factor determined in step 1.2.3.

c. Using the elapsed time from step 1.3.2, estimate a primary coolant leak rate by visually interpolating between the curves in Figure B.1. Record on worksheet 1-C the value of the specific activity from the curves above and below the normalized measured value.

d. If the reading from RM-90-106 only is off-scale, use Figures B.2-a, B.2-b, or B.2-c and the RM-90-112 reading. Then repeat steps 1.3.6 a through c.

e. Record on Worksheet 1-C the estimated primary coolant leak rate.

1.3.7 If the readings for both RM-90-106 and RM-90-112 are off-scale, do the following:

a. Select Figure B.3-a, B.3-b, or B.3-c depending on the isotope spectrum chosen.

b. Normalize the measured dose rate at monitor RM-90-2 location by dividing by the normalization factor determined in step 1.2.3.

c. Using the elapsed time from step 1.2.3, estimate a primary coolant leak rate by visually interpolating between the curves of Figure B.3. Record on worksheet 1-C the value of the calculated monitor reading from the curves above and below the normalized measured value.

d. Record on Worsheet 1-C, the estimated primary coolant leak rate.

1.4 Determine containment leak rate

Use worksheet 1-D. Record the reading of the shield building vent flow element FE-30-242. Record the time when the reading was taken. Also, record containment pressure and the time of reading.

1.4.1 Calculate the radioactivity release rate by multiplying the specific activity from RM-90-100 by the vent flow from FE-30-242 and by 472 (unit conversion factor):

Release Rate = 472 x Activity x Vent Flow

1.4.2 Select from Figure B.4 the sheet corresponding to the isotope spectrum and primary coolant leak rate determined earlier.

1.4.3 Divide this measured rate by the normalization factor from Step 1.2.3.

1.4.4 Using the elapsed time from step 1.3.2, estimate a gaseous leak rate by visually interpolating between the curves of Figure B.4. Record on Worksheet 1-D the value of the calculated vent release rate from the curves above and below the normalized measured value.

1.4.5 Record on Worksheet 1-D the estimated gaseous leak rate.

1.4.6 From Figure B.5, for the containment pressure determined in Step 1.4 and the calculated leak rate, obtain an equivalent hole size.

2.0 PREDICTED VENT RELEASE RATE--NO CHANGE IN PLANT CONDITIONS

2.1 Calculate noble gas vent release rate

Use worksheet 2-A.

2.1.1 Use the same sheet of Figure B.4 selected in Step 1.4.2.

2.1.2 Use the containment leak rate determined in Step 1.4.4.

2.1.3 Select the time for which the vent release rate is desired.

2.1.4 Obtain the noble gas vent release rate by interpolation between the curves on Figure B.4:

2.1.4.a Determine interpolation fraction = $\frac{LRD - LR(L)}{LR(U) - LR(L)}$

where: LRD = containment leak rate from Step 1.4.4.
LR = containment leak rate for which curve is calculated.
(U),(L) = value of LR immediately (above) (below) LRD.

2.1.4.b Determine the difference between vent release rates corresponding to LR(U) and LR(L) at the desired time, multiply by the interpolation fraction, and add to the vent release rate corresponding to LR(L).

2.1.4.c Multiply by the normalization factor from Step 1.2.2.

2.2 <u>Calculate the iodine vent release rate.</u>

Use worksheet 2-B.

2.2.1 Use the sheet of Figure B.6 corresponding to the one of Figure B.4 selected in Step 2.1.1.

2.2.2 Follow procedure 2.1. Substitute Figure B.6 for Figure B.4.

NOTE: It is not necessary to recalculate the interpolation fraction (Step 2.1.4a) since it will be unchanged.

3.0 <u>PREDICTED VENT RELEASE RATE--CHANGING PLANT CONDITIONS</u>

3.1 <u>For a different postulated primary coolant activity:</u>

3.1.1 Consider which isotope spectrum ("Expected," "Design," "Modified TID-14844") was chosen in Step 1.1.

3.1.2 Consider the magnitude of the normalization factor determined in Step 1.2.

3.1.3 Decide on a new primary coolant activity by increasing the normalization factor (fuel failure) or changing the isotope spectrum based on plausible developments in the condition of the plant. In order of increasing severity, isotope spectra rank as follows:

(1) Expected

(2) Design

(3) Modified TID-14844

If the isotope spectrum is "Expected," do not increase the normalization factor to more than 2; instead, change the spectrum to "Design."

If the isotope spectrum is "Design," do not increase the normalization factor to more than 15; instead, change the spectrum to "Modified TID-14844."

3.2 For a different postulated containment leak rate

3.2.1 For an anticipated change in containment pressure with no additional containment degradation, use Figure B.5 with the equivalent hole size determined in Step 1.4.5.

3.2.2 For anticipated containment degradation, use Figure B.5 with an appropriately chosen hole size.

3.2.3 Determine the new containment leak rate from Figure B.5.

3.3 For an anticipated degradation of the primary coolant loop.

Change the primary coolant leak rate to containment.

3.4 For the postulated changed plant parameters, recalculate future vent releases using the methods given in part 2.0.

NOTE: Part 2.0 is based on unchanged plant parameters. If the postulated conditions are more severe than those determined to exist in part 1.0, the part 2.0 methods with the changed parameters will result in release rates which are too high; conversely, in the unlikely event that the postulated conditions are less severe, the estimated release rates will be too low.

GASEOUS RELEASES WORKSHEET
1-A

1.1 Determination of Isotopic spectrum

1.1.1 I_m = I-131 specific activity __________ μCi/g
C_m = Cs-137 specific activity __________ μCi/g
R_m = Rb-88 specific activity __________ μCi/g

I_m/C_m = __________ I_m/R_m = __________ C_m/R_m = __________

1.1.2

TABLE 1.1
Selection of Isotope Spectrum

I/R	I/R	C/R	Appropriate Isotope Spectrum
8.5 · I/C 18.5	1.0 I/R 10	C/R 0.18	Expected
I/C 8.5	I/R 1.0	0.18 C/R 1.0	Design
I/C 18.5	I/R 10	C/R 1.0	Modified TID-14844

1.1.3 Chosen distribution is (check one):

____ "Expected"

____ "Design"

____ "Modified TID-14844"

1.2 Determination of severity of fuel failure:

1.2.1

TABLE 1.2
Specific Activity in Primary Coolant (Prior to Incident) for Three Isotopic Spectra

Isotope	Specific Activity Level (μCi/g) Expected	Design	Modified TID-14844
I_c = I-131	2.3	2.5	3.57×10^3
C_c = Cs-137	0.16	1.0	1.59×10^2
R_c = Rb-88	1.9	3.7	9.09×10^1

GASEOUS RELEASES WORKSHEET
1-A (Continued)

1.2.2 Subscript m denotes measured specific activity.
Subscript c denotes calculated specific activity from Table 2.

I_m/I_c = ________ C_m/C_c = ________ R_m/R_c = ________

1.2.3 Select a normalization factor corresponding most closely to two of the three ratios:

Normalization Factor F = ____

GASEOUS RELEASES WORKSHEET
1-B

1.3 Radiation Monitor Readings at ________________________
Time and Date

RM-90-100: _________ cpm = _________ μCi/cc
RM-90-112: _________ cpm = _________ μCi/cc
RM-90-106: _________ cpm = _________ μCi/cc

RM-90-2 _________ mr/hr

1.3.2 Reactor shutdown date and time _______________

Monitor reading date and time _______________

Elapsed time t_e = _______ hr _______ min = _______ hrs

1.3.4 Detector Efficiency Verification Date and Time: _________

RM-90-112: Xe-133 _______ cpm = _______ μCi/cc
(Upper Containment) Total _______ cpm = _______ μCi/cc

Ratio = ______________

RM-90-106: Xe-133 _______ cpm = _______ μCi/cc
(Lower Containment) Total _______ cpm = _______ μCi/cc

Ratio = ______________

1.3.5 Measured containment Xe-133 specific activity:

Xe-133 in Upper Containment:

___________ μCi/cc x ______ = _________ μCi/cc =MXe
(1-RM-90-112) (Ratio)

Xe-133 in Lower Containment:

___________ μCi/cc x ______ = _________ μCi/cc =MXe
(1-RM-90-106) (Ratio)

GASEOUS RELEASES WORKSHEET
1-C

1.3.6 Calculate primary coolant leak rate using monitor RM-90-106 or RM-90-112 from Figure B.1-a (lower containment, monitor RM-90-106) or Figure B.2-a (upper containment, monitor RM-90-112) for "Expected," Figure B.1-b or B.2-b for "Design," or Figure B.1-c or B.2-c for "Modified TID-14844."

1.3.6.b From 1.3.5 worksheet 1-B.

MXe = ________ μCi/cc (RM-90-106) or MXE = ________ μCi/cc (RM-90-112)

where MXe is the measured Xe-133 specific activity at t_e.

Normalization factor (From 1.2.3, worksheet 1-A) F = ________

MXeN = MXe/F: ________ μCi/cc (RM-90-106) or ________ μCi/cc (RM-90-112)

1.3.6.c Calculated specific activity from Figure _____ (Record figure used.)

CXe(U) = ________ μCi/cc CXe(L) = ________ μCi/cc

where: CXe(U) = Calculated Xe-133 activity greater than MXeN
CXe(L) = Calculated Xe-133 activity smaller than MXeN

1.3.6.e Primary coolant leak rate: ________ gpm

1.3.7 Calculate primary coolant leak rate using monitor RM-90-2:

1.3.7.b From 1.3, worksheet 1-B, RM-90-2

MDR = ________ mr/h

where MDR is the measured dose rate at t_e

Normalization factor (from 1.2.3, worksheet 1-A) F = ________

MDRN = MDR/F: ________ mr/h

1.3.7.c Calculated monitor reading from Figure B.3 _____ (Record figure used.)

CDR(U) = ________ mr/h CDR(L) = ________ mr/h

where: CDR(U) = Calculated monitor reading greater than MDRN
CDR(L) = Calculated monitor reading smaller than MDRN

1.3.7.d Primary coolant leak rate: ________ gpm

GASEOUS RELEASES WORKSHEET
1-D

1.4 Flow Element FE-30-242: _______ cfm at ____________________
Time and Date

			Time and Date
Containment Pressure PdI-30-42:	_______	psi	____________
PdI-30-43:	_______	psi	____________
PdI-30-44:	_______	psi	____________
PdI-30-45:	_______	psi	____________

From worksheet 1-B.

RM-90-100: __________ µCi/cc

1.4.1 Measured Shield Building Vent Release Rate:

VRR = 472 x __________ (RM-90-100) x _________ (FE-30-242) = __________ µCi/s

1.4.3 Normalization factor (from 1.2.3, worksheet 1-A) F = _______

VRRN = VRR/F: ____________ µCi/s

1.4.4 Calculated vent release rate from Figure B.4 _____ (Record figure used.)

CRR(U) ___________ µCi/s CRR(L) ___________ µCi/s

where: CRR(U) = Calculated vent release rate greater than VRRN
CRR(L) = Calculated vent release rate smaller than VRRN

1.4.5 Containment gaseous leak rate:

LRD = _______________ scfh

1.4.6 Equivalent hole size diameter _________ in.

GASEOUS RELEASES WORKSHEET
2-A

Containment Noble Gas Release Rate

Containment Leak Rate From Worksheet 1-D: LRD = _______ scfh

2.1.2 From Figure B.4 _______ (Record figure used.)

LRU (U) = _______ scfh LR(L) = _______ scfh

where: LR = Containment leak rate used in Step 1.4.4

2.1.3 Time elapsed since reactor shutdown t: _________ hours

2.1.4 Interpolation Fraction:

$$IF = \frac{LRD - LR(L)}{LR(U) - LR(L)}$$

$$= \frac{(\qquad) - (\qquad)}{(\qquad) - (\qquad)} = \underline{\qquad\qquad\qquad}$$

From Figure B.4, at t, determine CRR(U) corresponding to LR(U):

CRR(U) = _________ μCi/s

From Figure B.4, at t, determine CRR(L) corresponding to LR(L):

CRR(L) = _________ μ Ci/s

Normalization factor (from 1.2.3, worksheet 1-A) F = ________

Future noble gas release rate =

$$[IF \times (CRR(U) - CRR(L)) + CRR(L)] \times F$$

$$= [\underline{\qquad} \times (\underline{\qquad} - \underline{\qquad}) + \underline{\qquad}] \times \underline{\qquad}$$

= _______ μCi/s

NOTE: Same nomenclature as on worksheet 1-D.

GASEOUS RELEASES WORKSHEET
2-B

Interpolation Fraction from Worksheet 2-A: IF = ________

2.2.1 From Figure B.6 ____: (Same nomenclature as worksheet 2-A.)

CRR(L) = ______ μCi/s   CRR(U) = ______ μCi/s

Normalization Factor F = ____________

Future iodine release rate:

[IF x (CRR(U) - CRR(L)) + CRR(L)] x F

= [______ x (______ - ______) + ______] x ______

= ______ μCi/s

FIGURES

FIGURES B.1

FIGURES B.2

FIGURES B.3

FIGURES B.4

FIGURE B.5

B.5 Containment leak rate versus pressure for varying hole sizes

FIGURES B.6

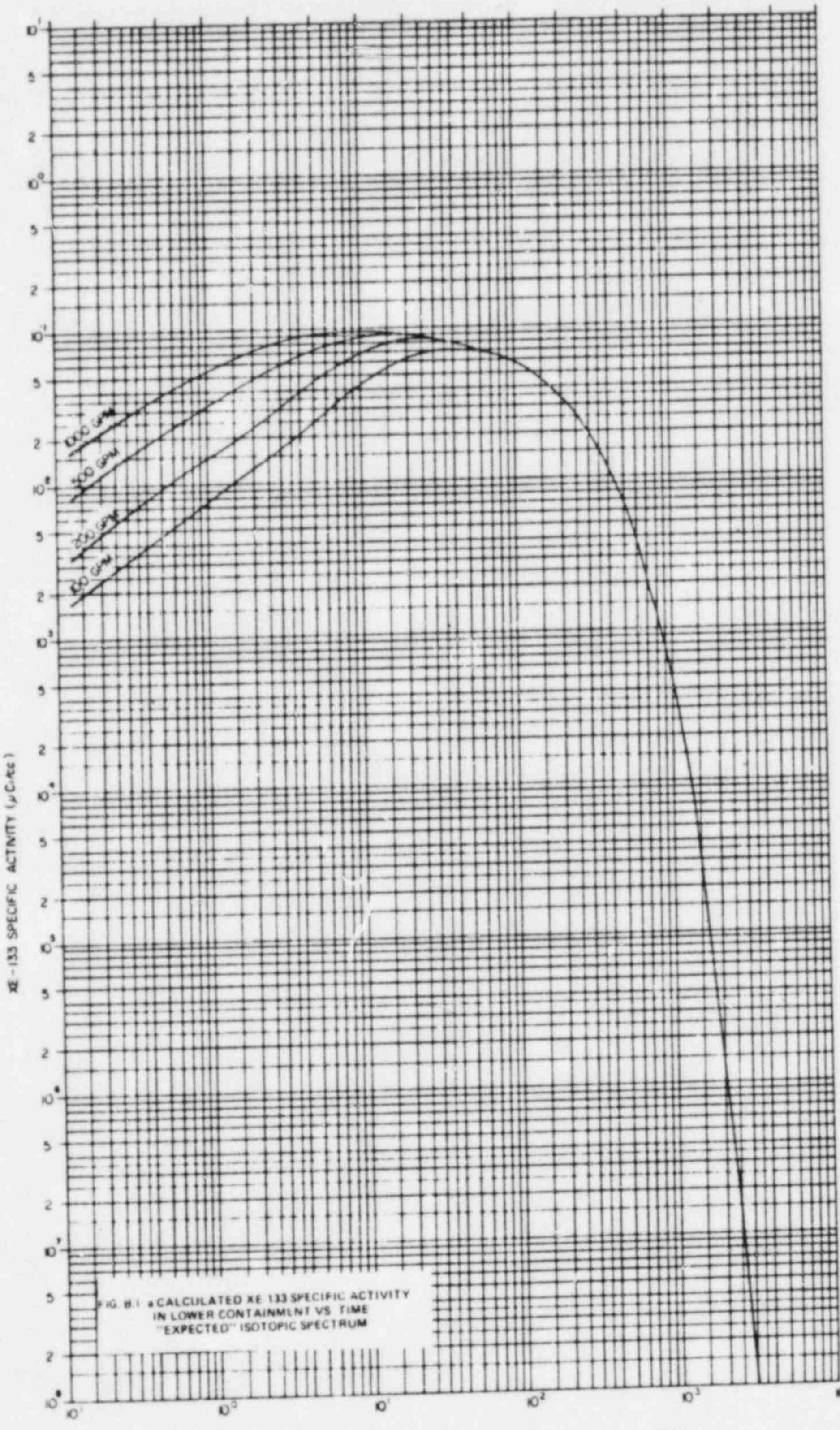


FIG. B.1-a CALCULATED XE 133 SPECIFIC ACTIVITY IN LOWER CONTAINMENT VS TIME "EXPECTED" ISOTOPIC SPECTRUM

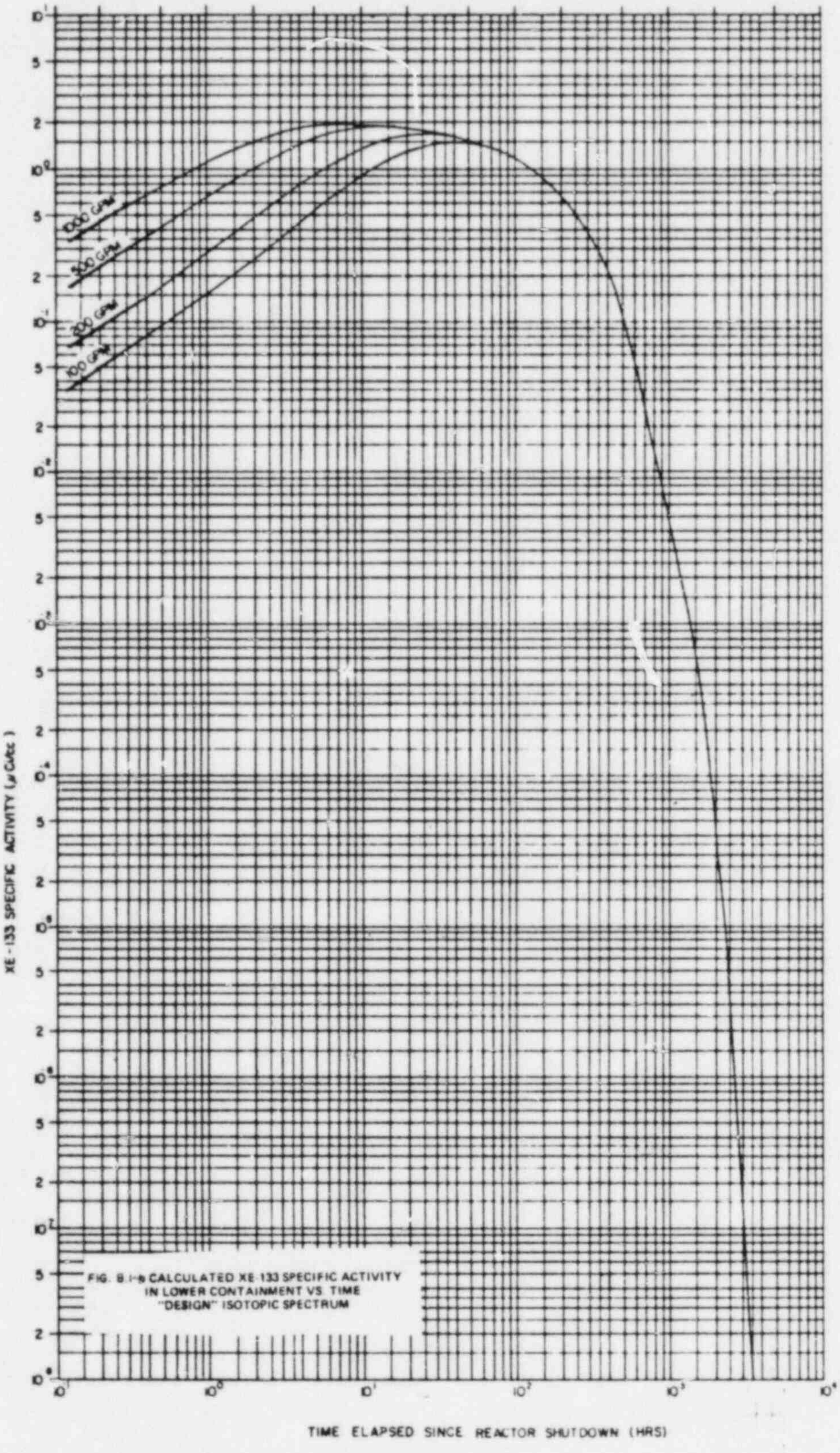


FIG. B.1-b CALCULATED XE-133 SPECIFIC ACTIVITY IN LOWER CONTAINMENT VS. TIME "DESIGN" ISOTOPIC SPECTRUM

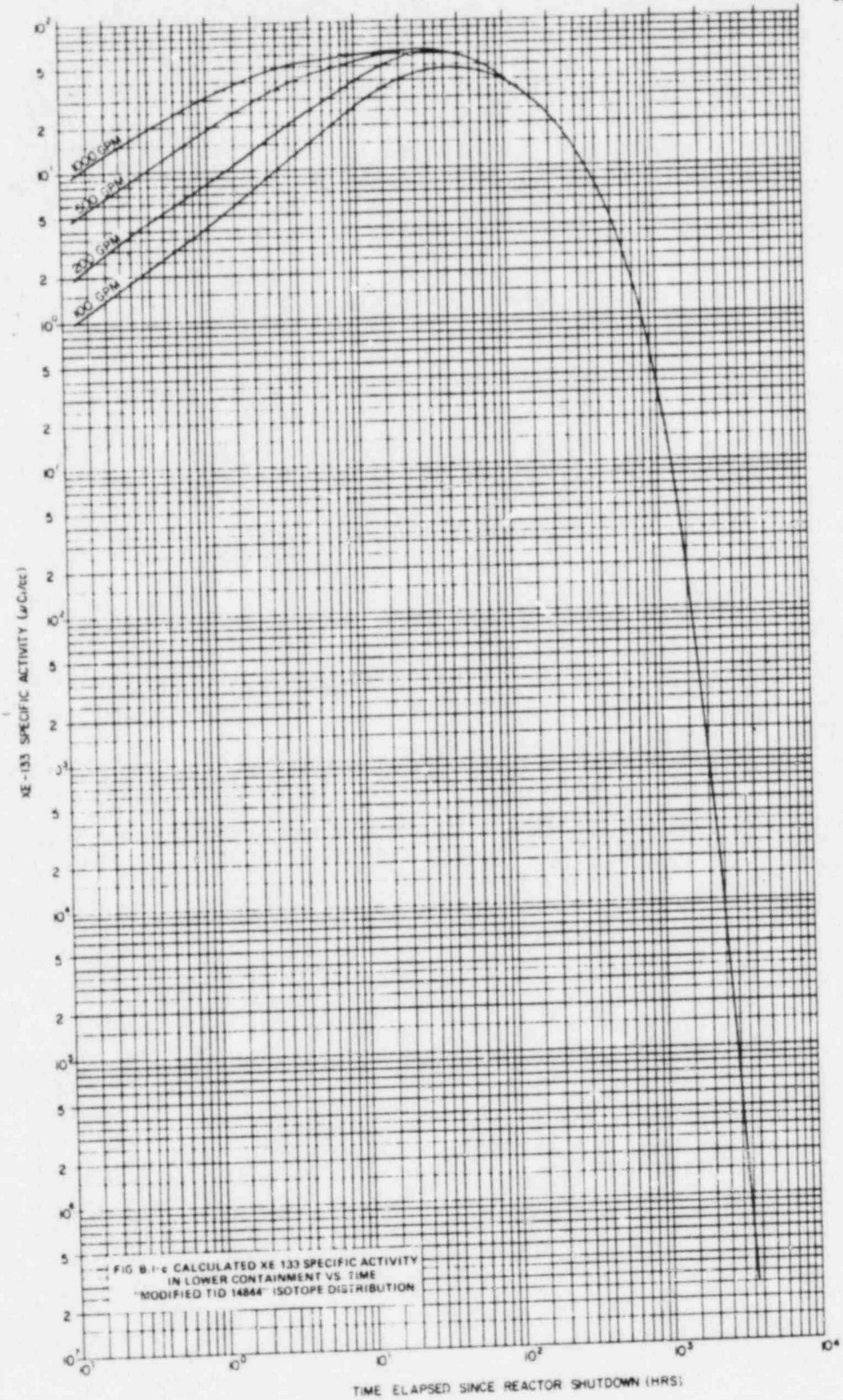


FIG B.1-c CALCULATED XE 133 SPECIFIC ACTIVITY IN LOWER CONTAINMENT VS. TIME "MODIFIED TID 14844" ISOTOPE DISTRIBUTION

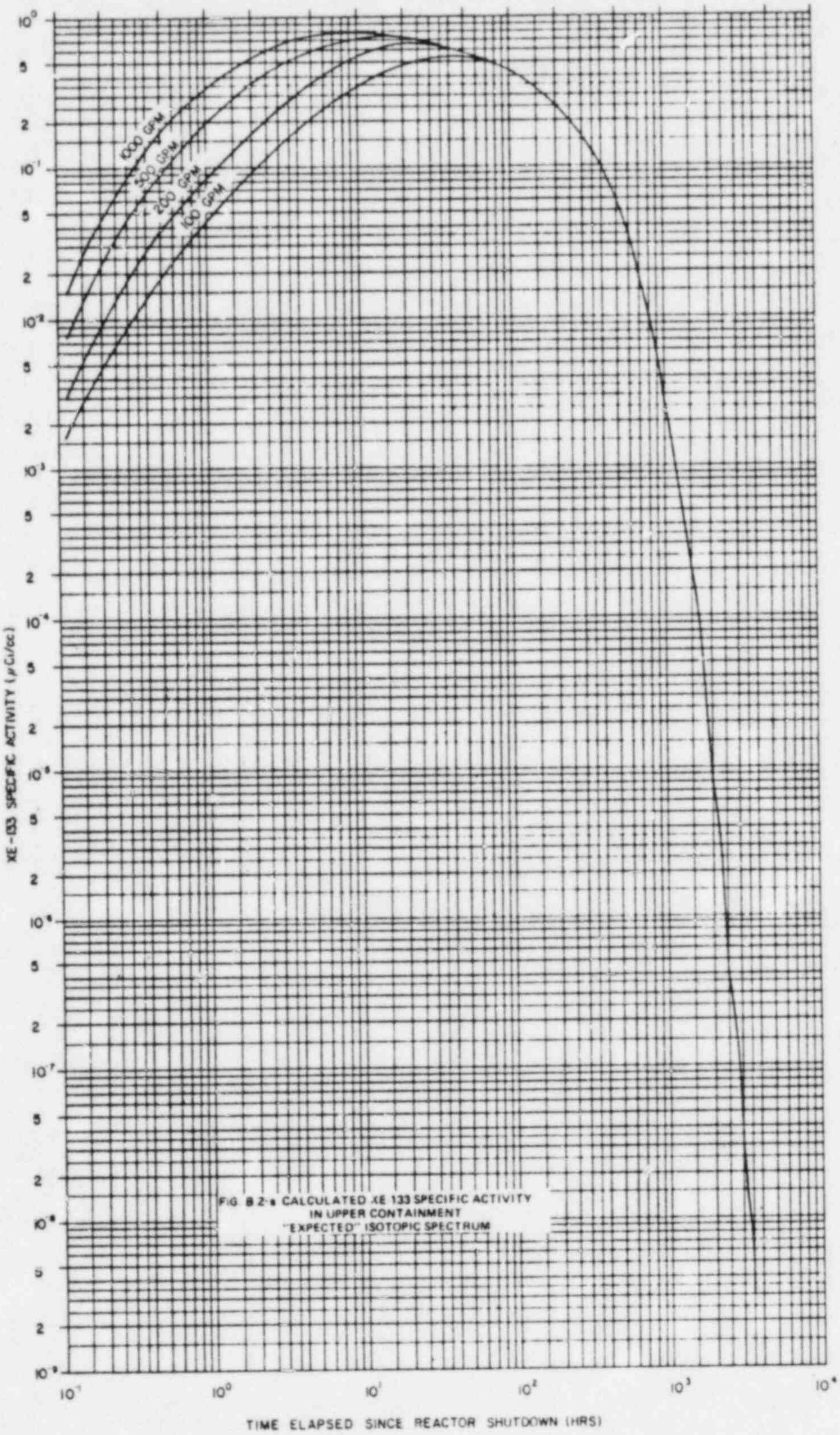


FIG. B.2-a CALCULATED XE 133 SPECIFIC ACTIVITY IN UPPER CONTAINMENT "EXPECTED" ISOTOPIC SPECTRUM

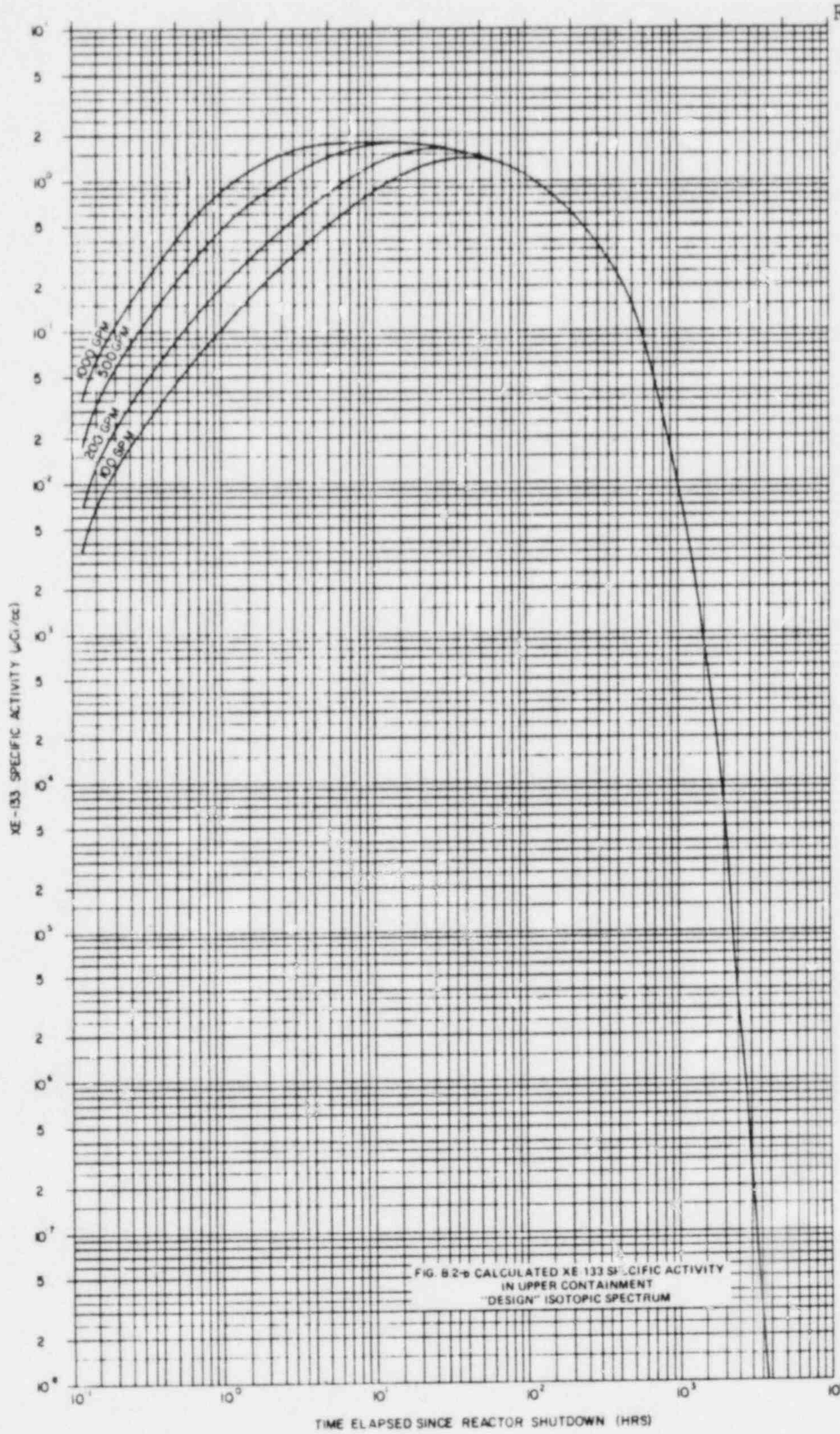


FIG. B.2-b CALCULATED XE 133 SPECIFIC ACTIVITY IN UPPER CONTAINMENT "DESIGN" ISOTOPIC SPECTRUM

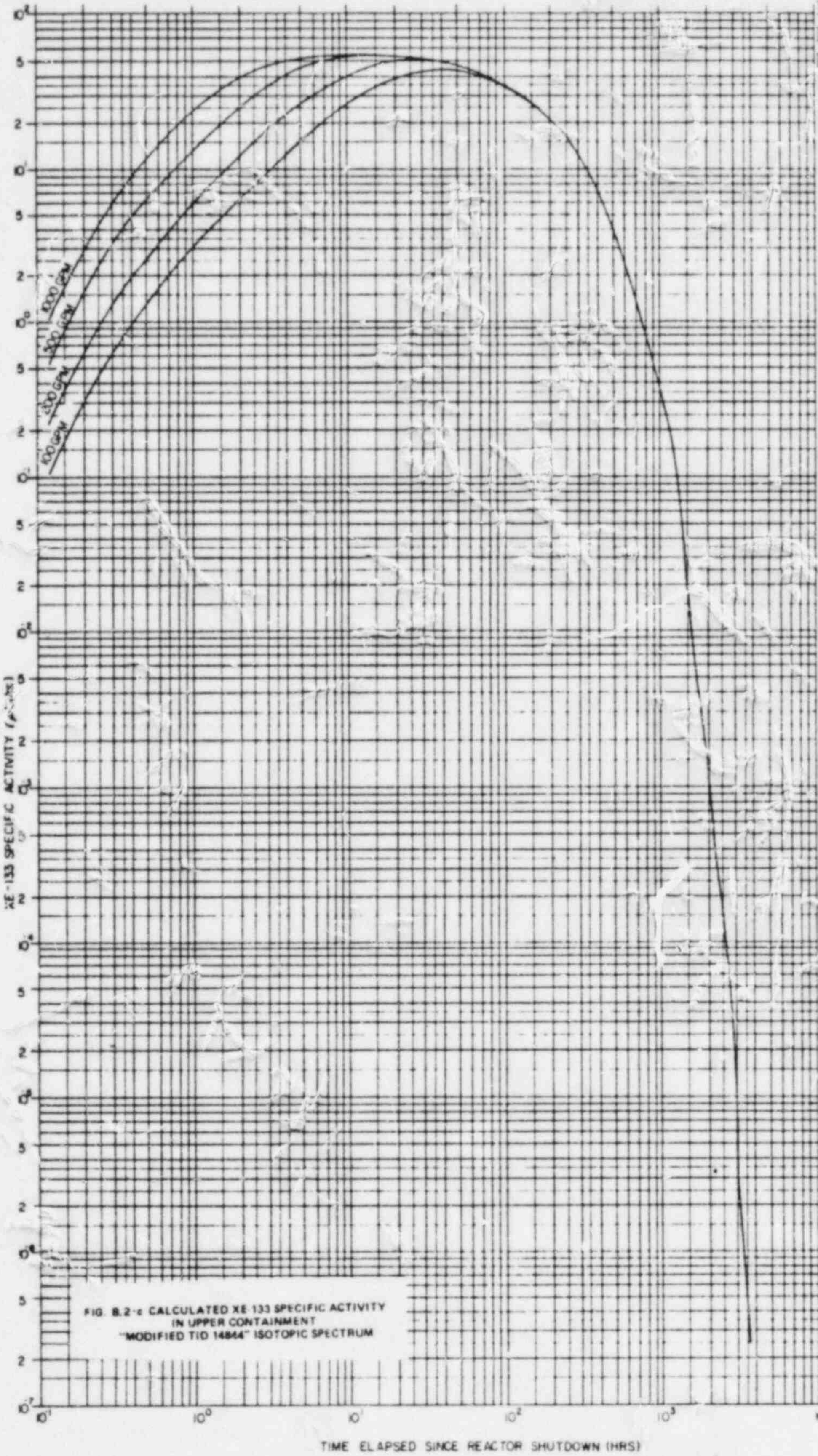


FIG. B.2-c CALCULATED XE 133 SPECIFIC ACTIVITY IN UPPER CONTAINMENT "MODIFIED TID 14844" ISOTOPIC SPECTRUM

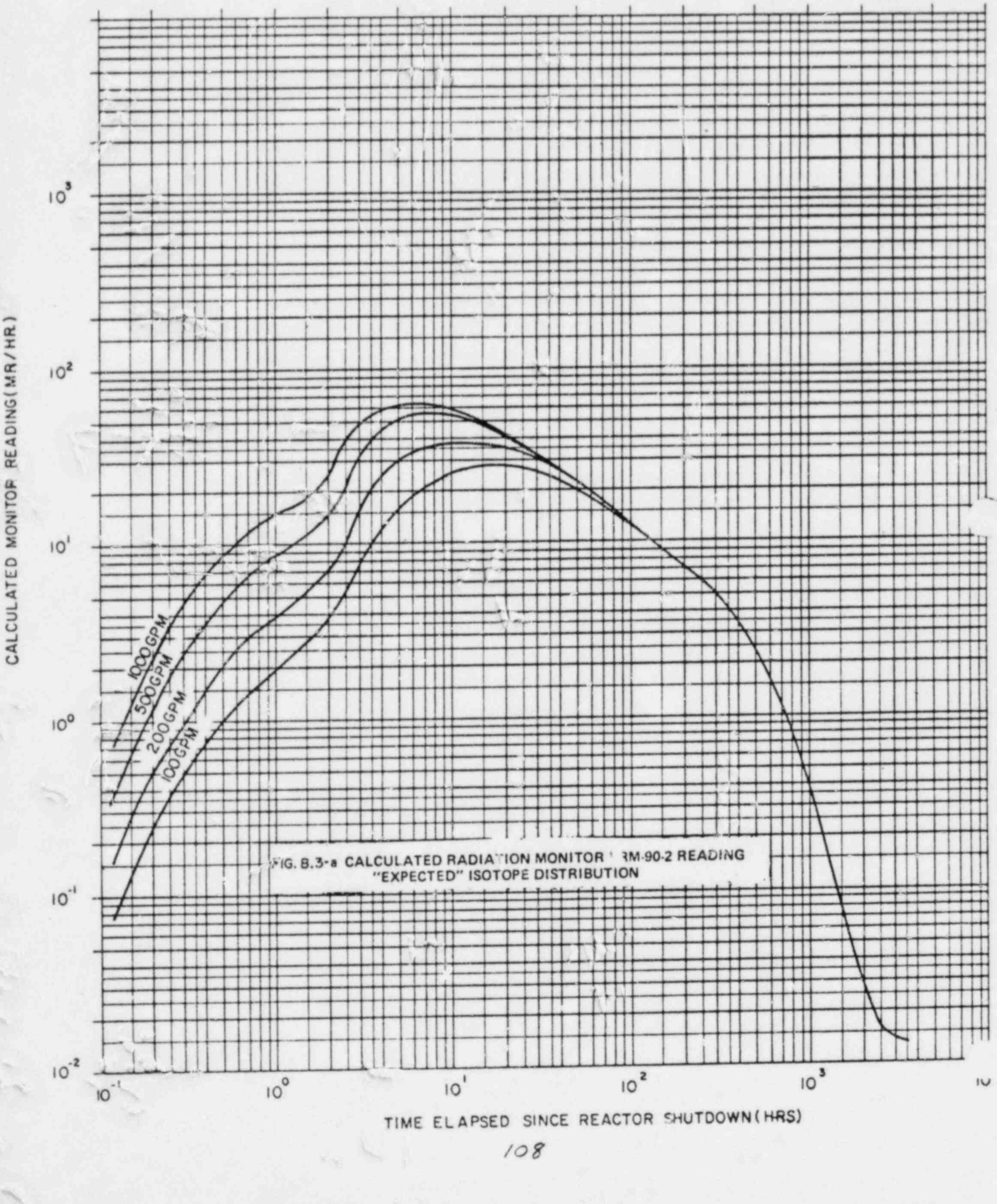


FIG. B.3-a CALCULATED RADIATION MONITOR RM-90-2 READING "EXPECTED" ISOTOPE DISTRIBUTION

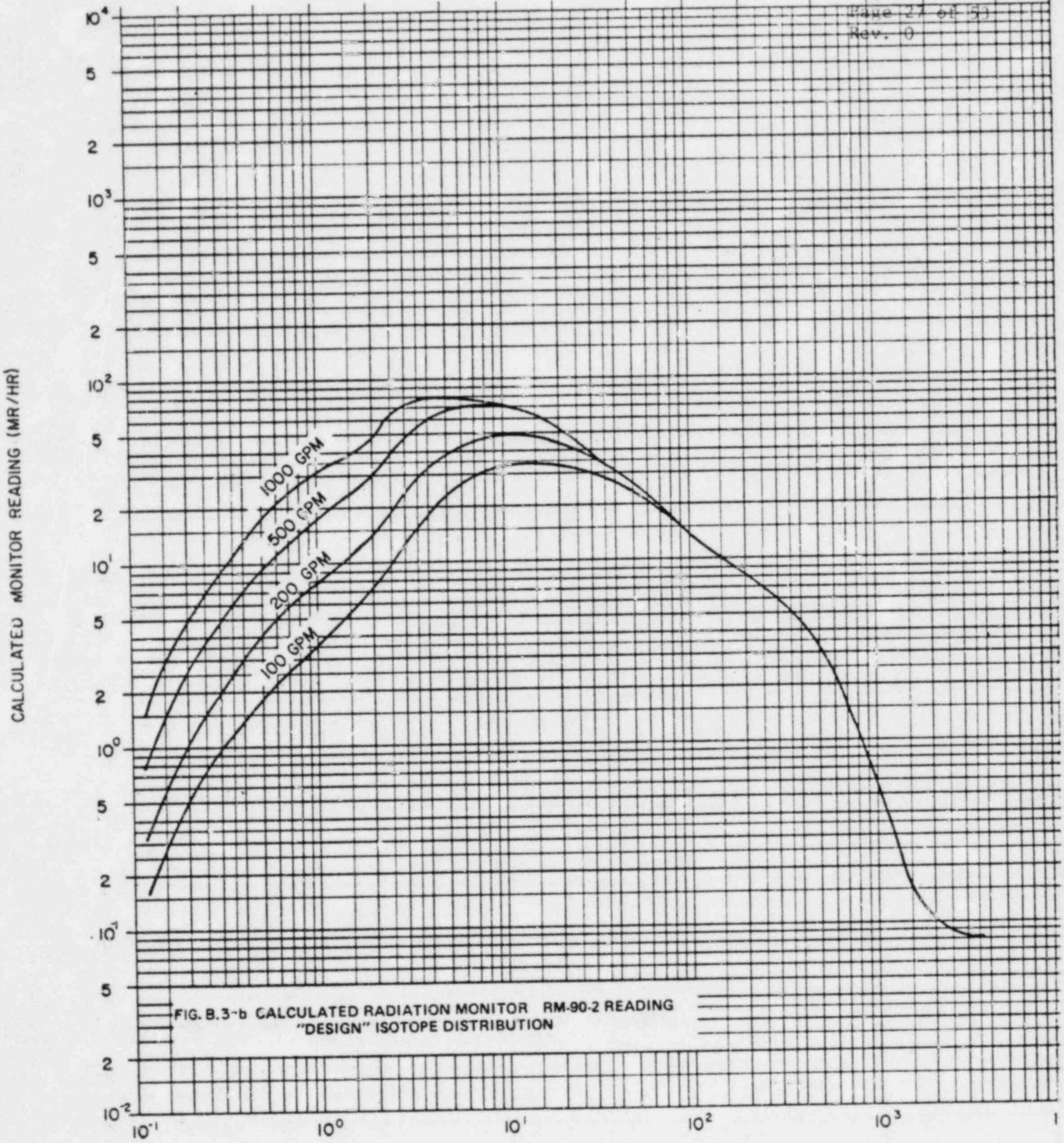


FIG. B.3-b CALCULATED RADIATION MONITOR RM-90-2 READING "DESIGN" ISOTOPE DISTRIBUTION

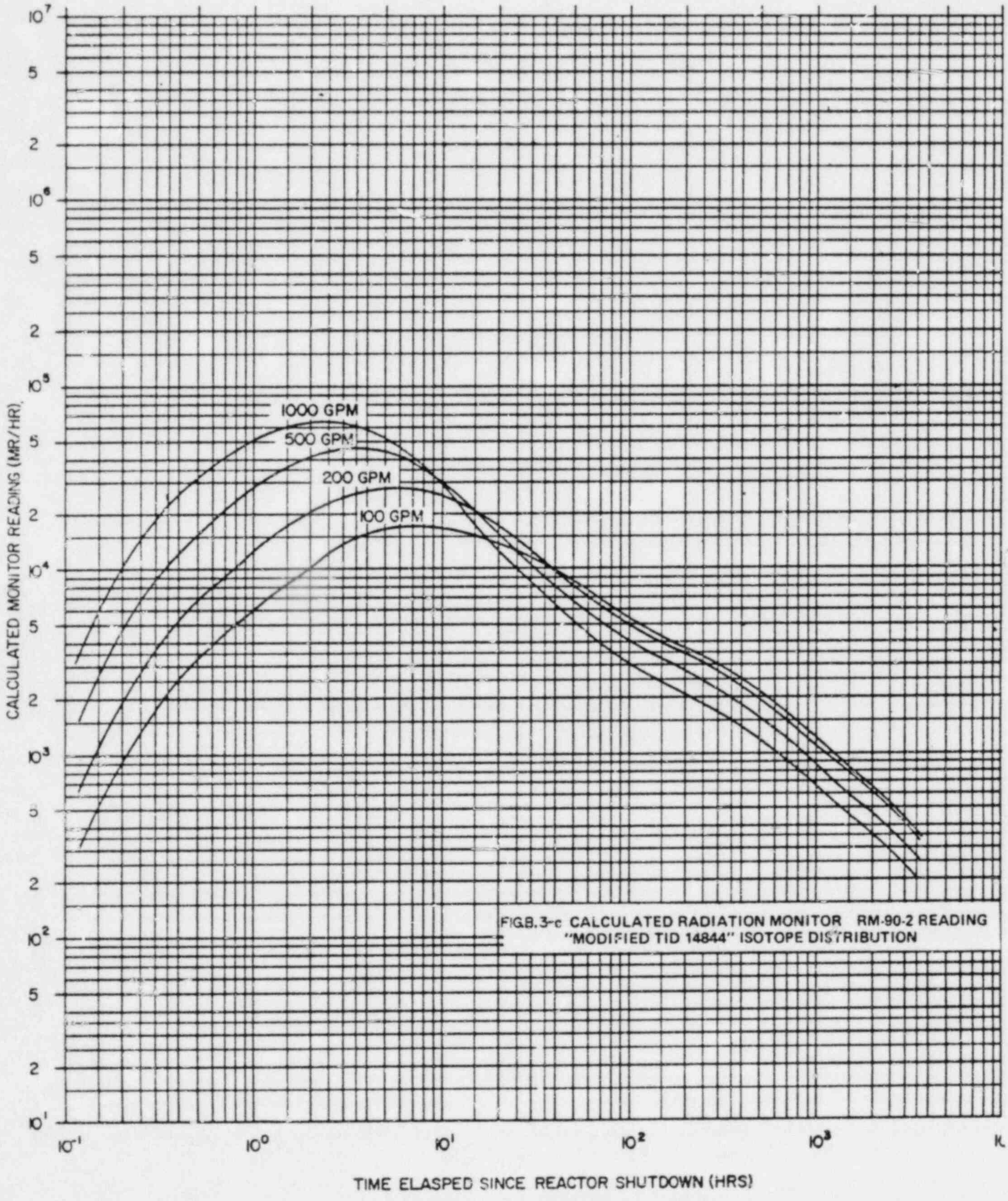


FIG.B.3-c CALCULATED RADIATION MONITOR RM-90-2 READING
"MODIFIED TID 14844" ISOTOPE DISTRIBUTION

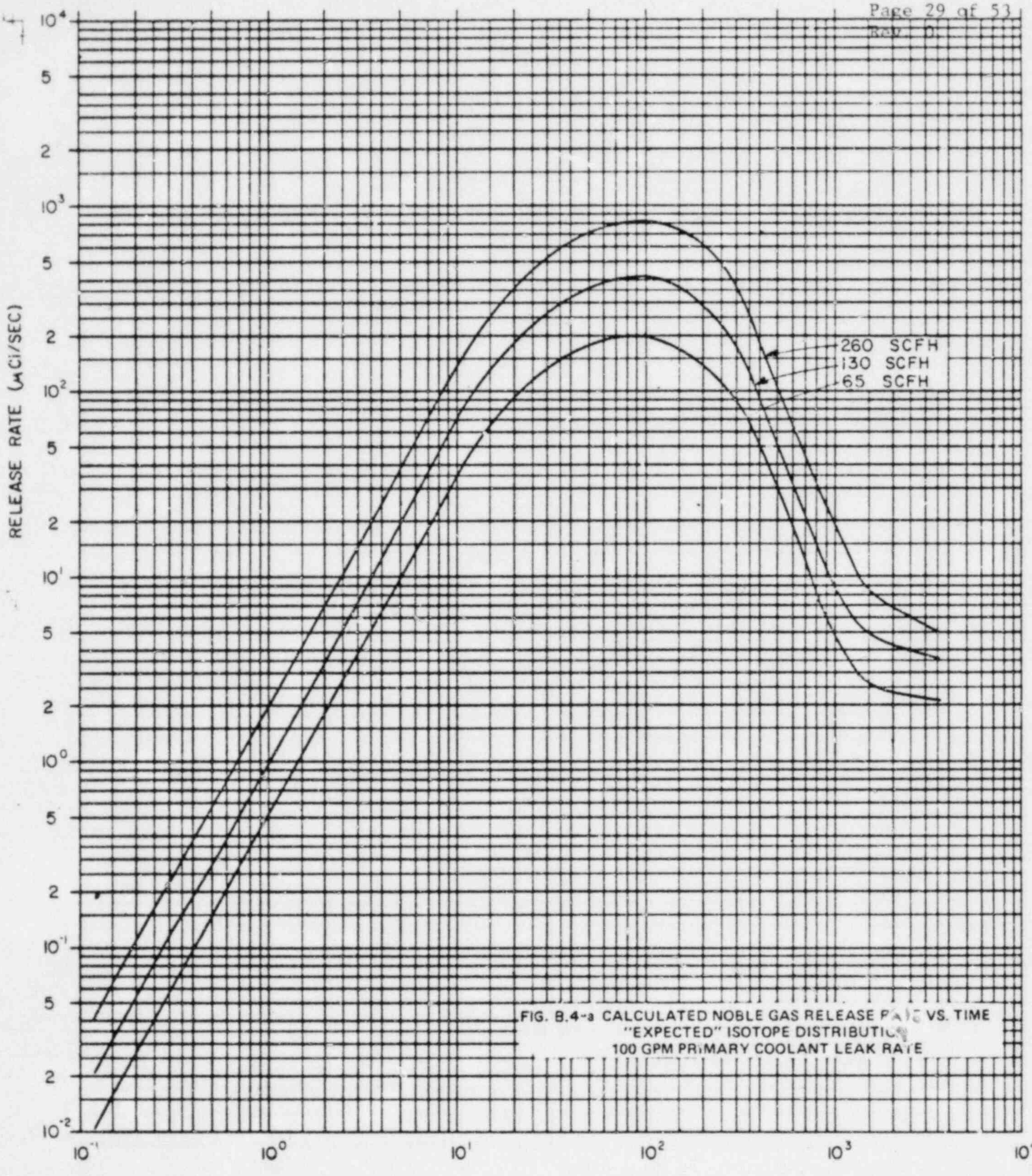


FIG. B.4-a CALCULATED NOBLE GAS RELEASE RATE VS. TIME
"EXPECTED" ISOTOPE DISTRIBUTION
100 GPM PRIMARY COOLANT LEAK RATE

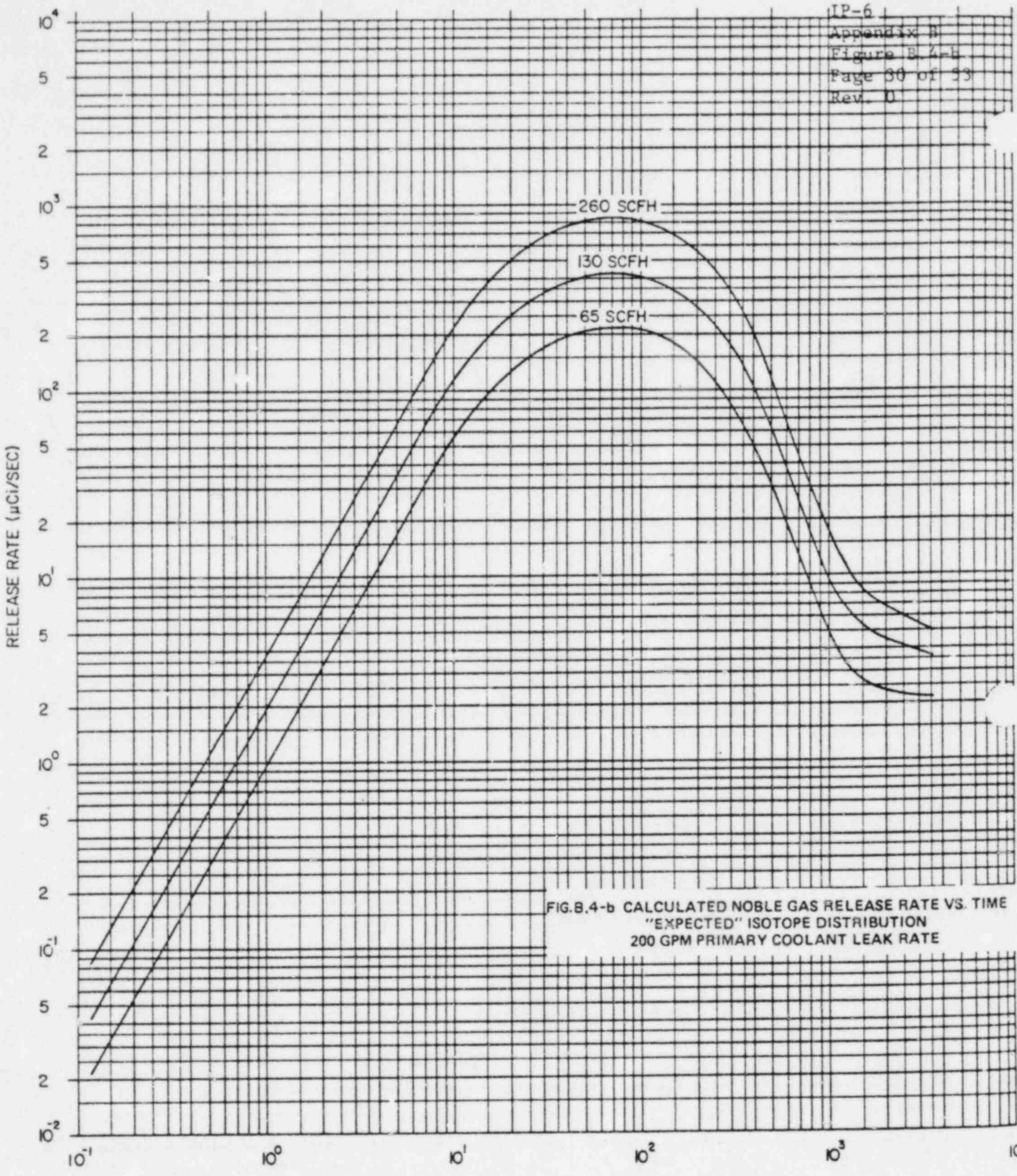


FIG. B.4-b CALCULATED NOBLE GAS RELEASE RATE VS. TIME
"EXPECTED" ISOTOPE DISTRIBUTION
200 GPM PRIMARY COOLANT LEAK RATE

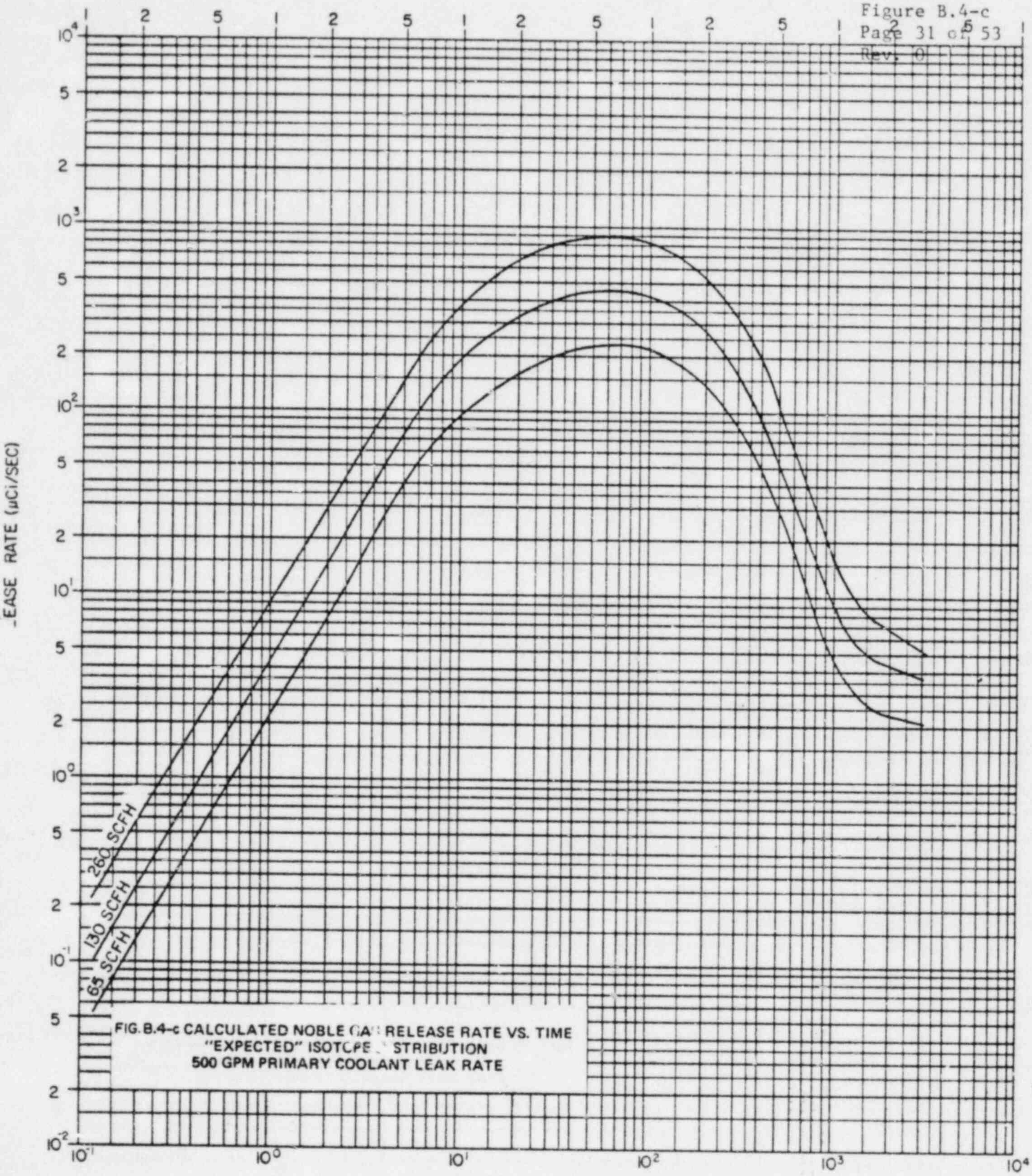


FIG. B.4-c CALCULATED NOBLE GAS RELEASE RATE VS. TIME
"EXPECTED" ISOTOPE DISTRIBUTION
500 GPM PRIMARY COOLANT LEAK RATE

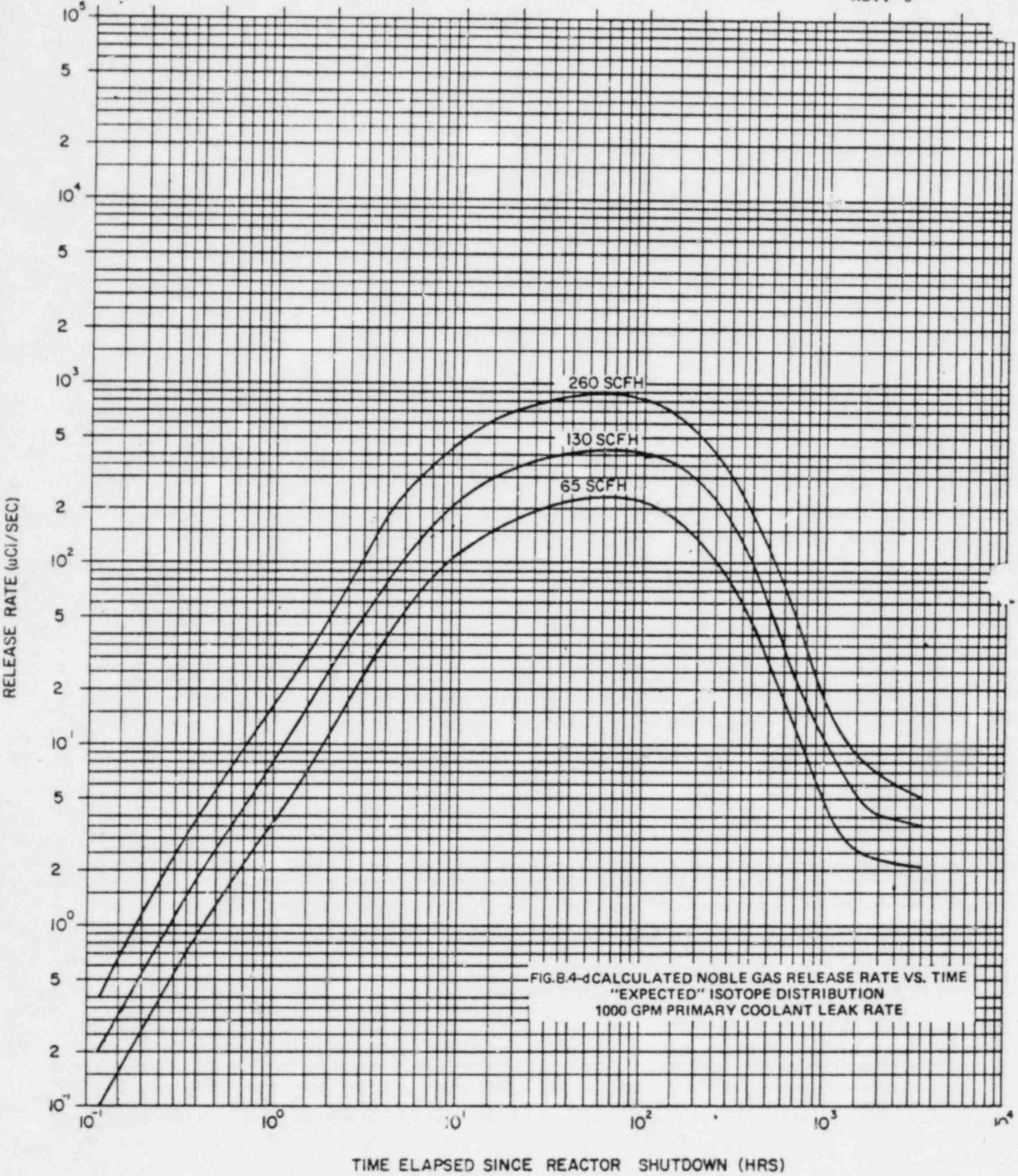


FIG.B.4-d CALCULATED NOBLE GAS RELEASE RATE VS. TIME
"EXPECTED" ISOTOPE DISTRIBUTION
1000 GPM PRIMARY COOLANT LEAK RATE

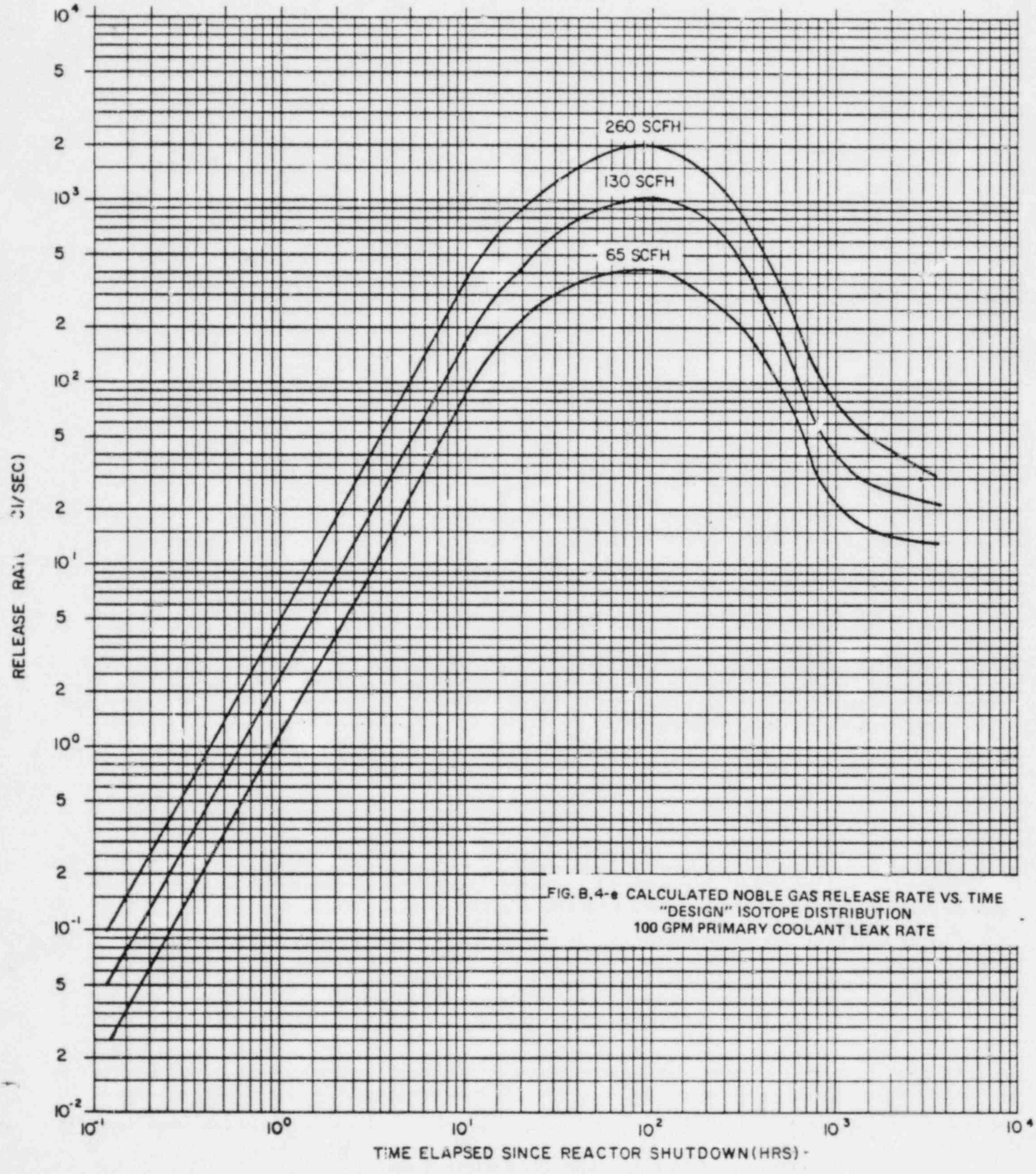


FIG. B.4-e CALCULATED NOBLE GAS RELEASE RATE VS. TIME
"DESIGN" ISOTOPE DISTRIBUTION
100 GPM PRIMARY COOLANT LEAK RATE

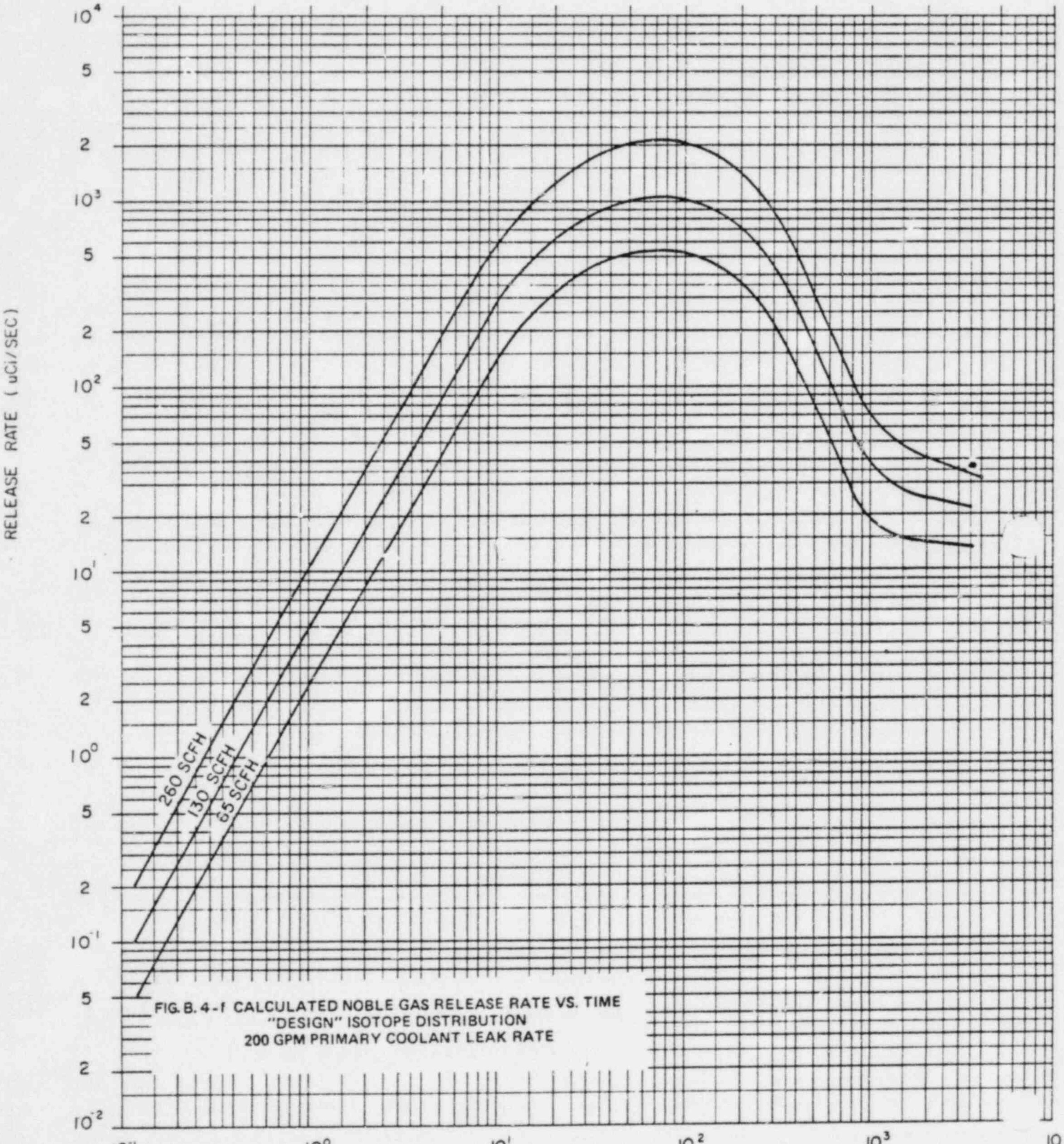


FIG. B. 4-f CALCULATED NOBLE GAS RELEASE RATE VS. TIME
"DESIGN" ISOTOPE DISTRIBUTION
200 GPM PRIMARY COOLANT LEAK RATE

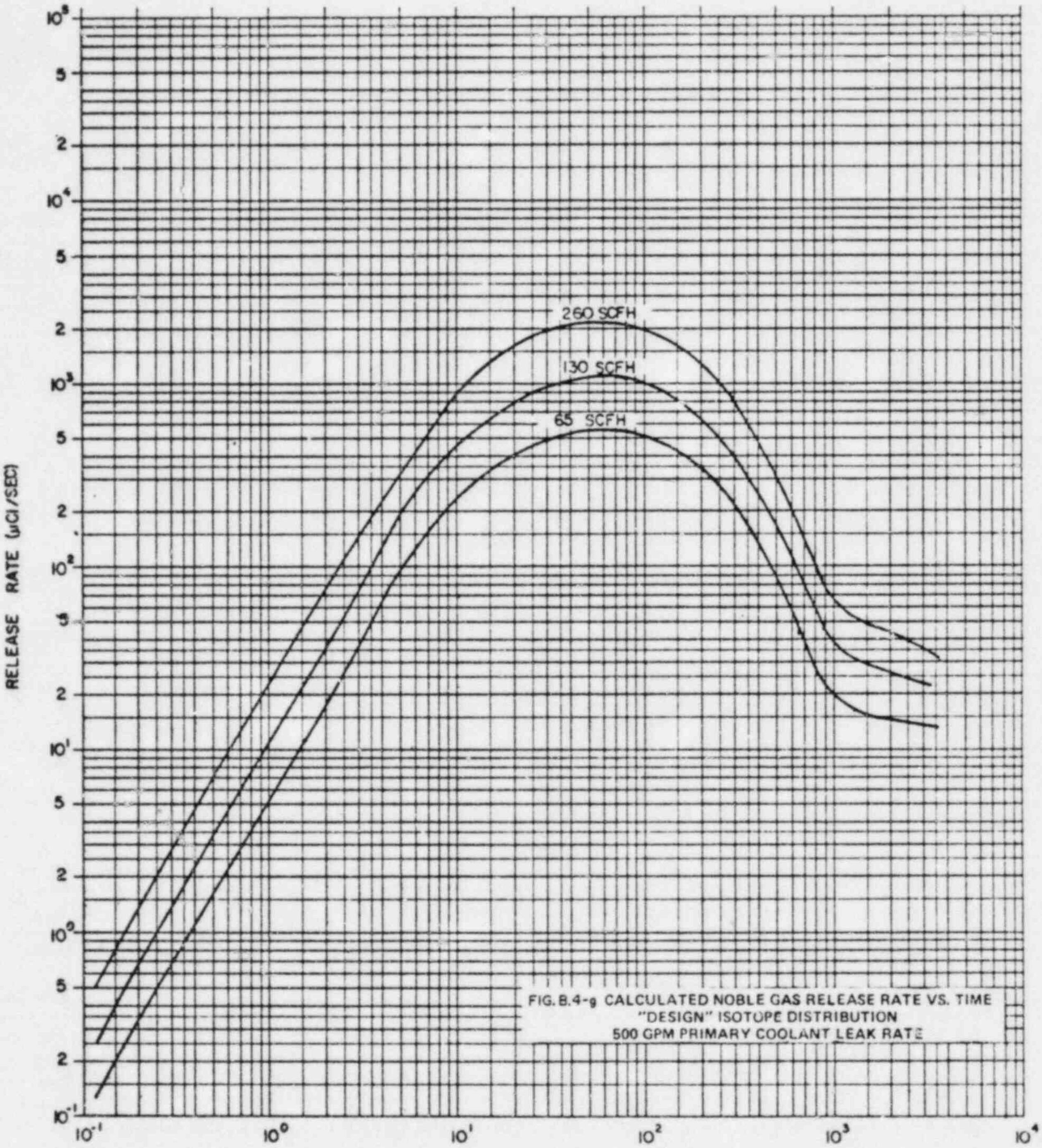


FIG. B.4-g CALCULATED NOBLE GAS RELEASE RATE VS. TIME
"DESIGN" ISOTOPE DISTRIBUTION
500 GPM PRIMARY COOLANT LEAK RATE

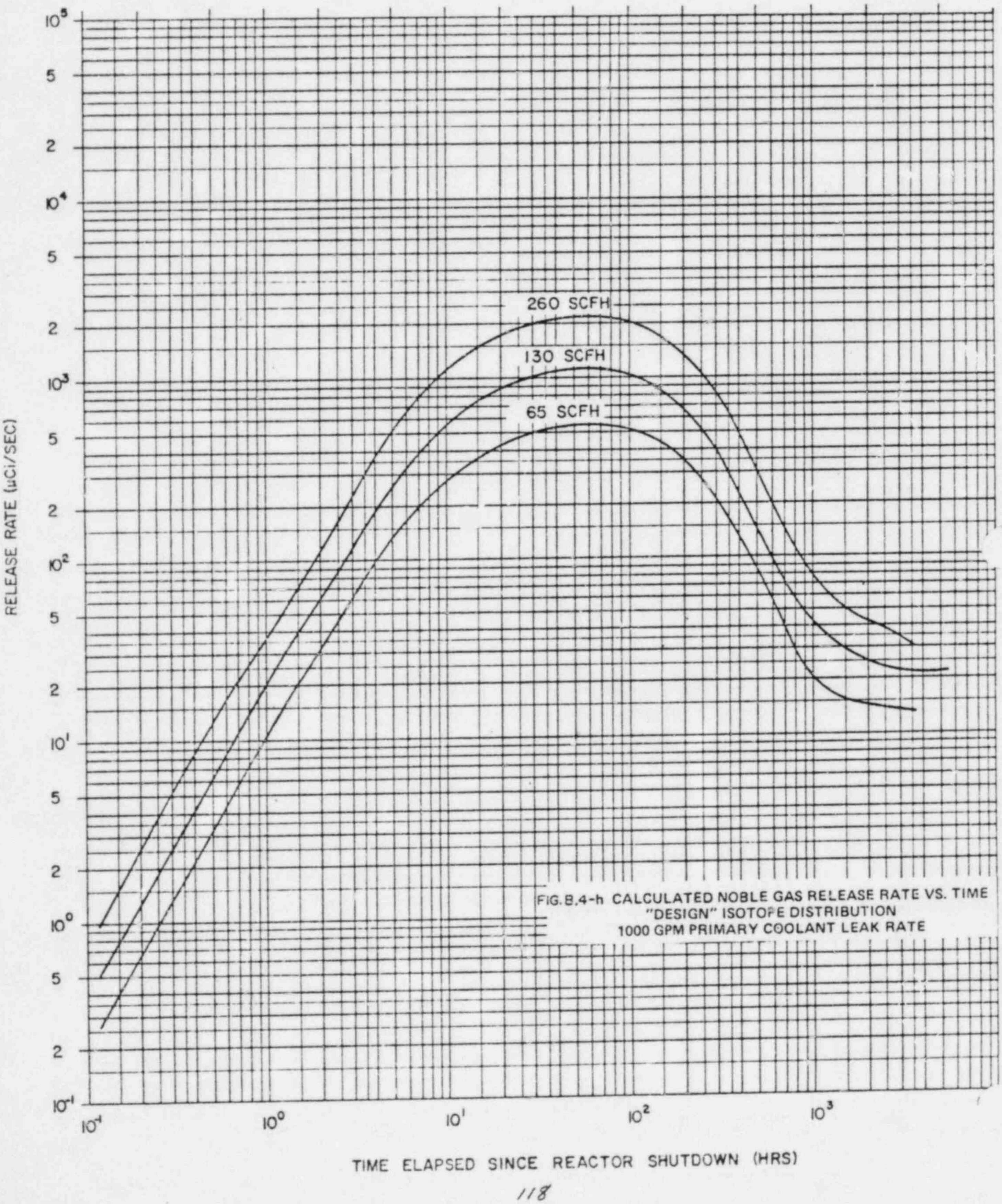


FIG. B.4-h CALCULATED NOBLE GAS RELEASE RATE VS. TIME
"DESIGN" ISOTOPE DISTRIBUTION
1000 GPM PRIMARY COOLANT LEAK RATE

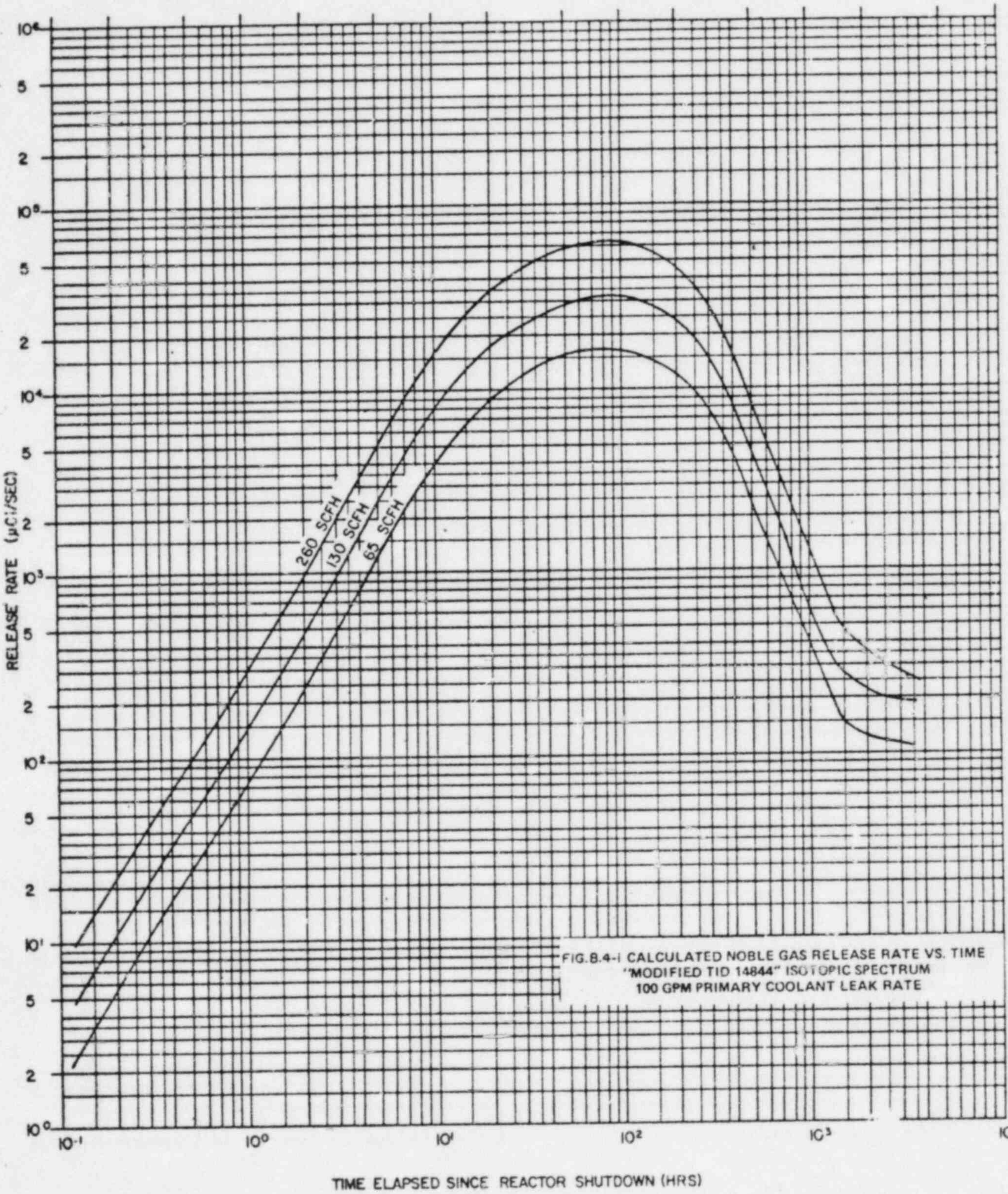


FIG.B.4-I CALCULATED NOBLE GAS RELEASE RATE VS. TIME
"MODIFIED TID 14844" ISOTOPIC SPECTRUM
100 GPM PRIMARY COOLANT LEAK RATE

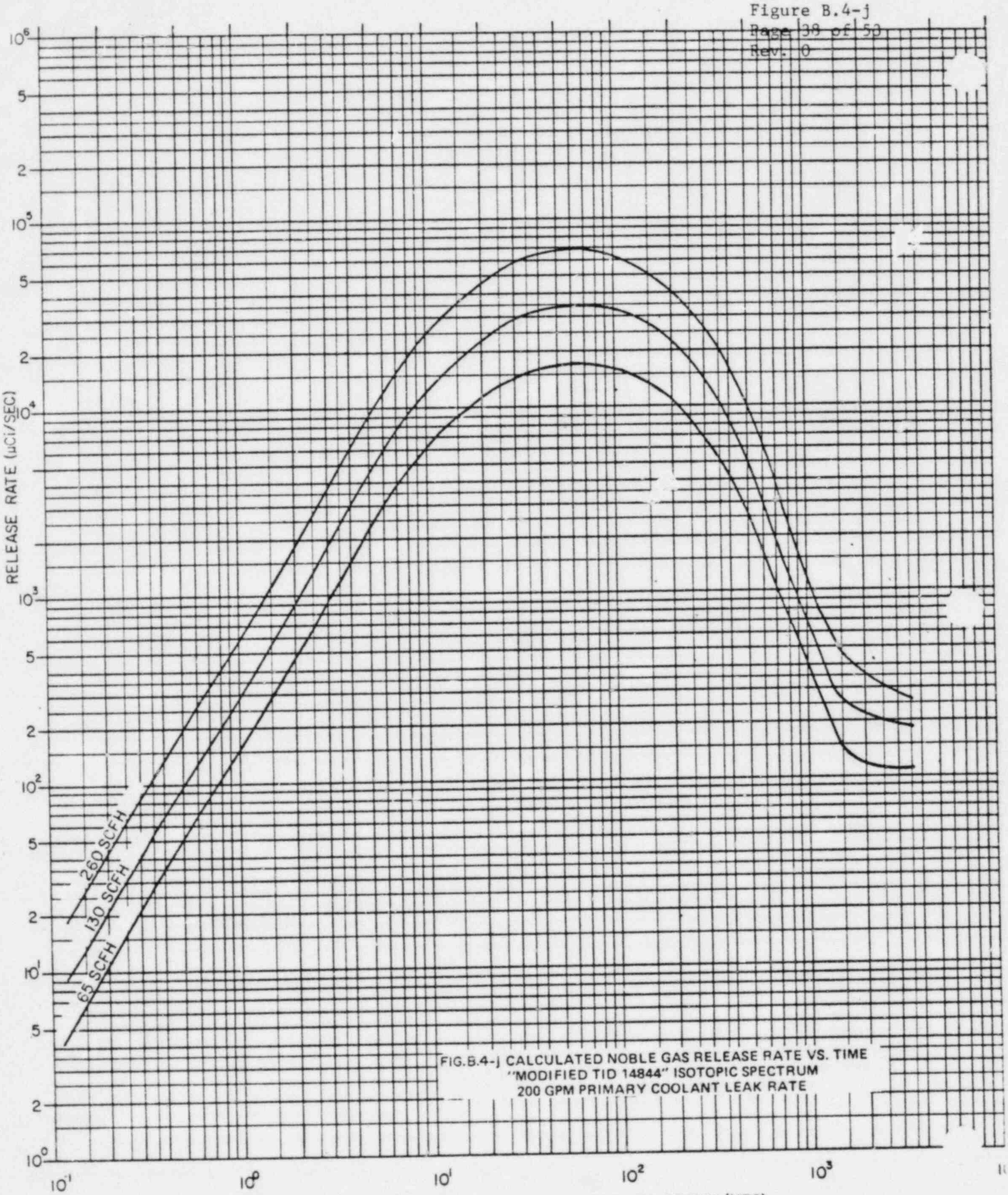


FIG.B.4-j CALCULATED NOBLE GAS RELEASE RATE VS. TIME
"MODIFIED TID 14844" ISOTOPIC SPECTRUM
200 GPM PRIMARY COOLANT LEAK RATE

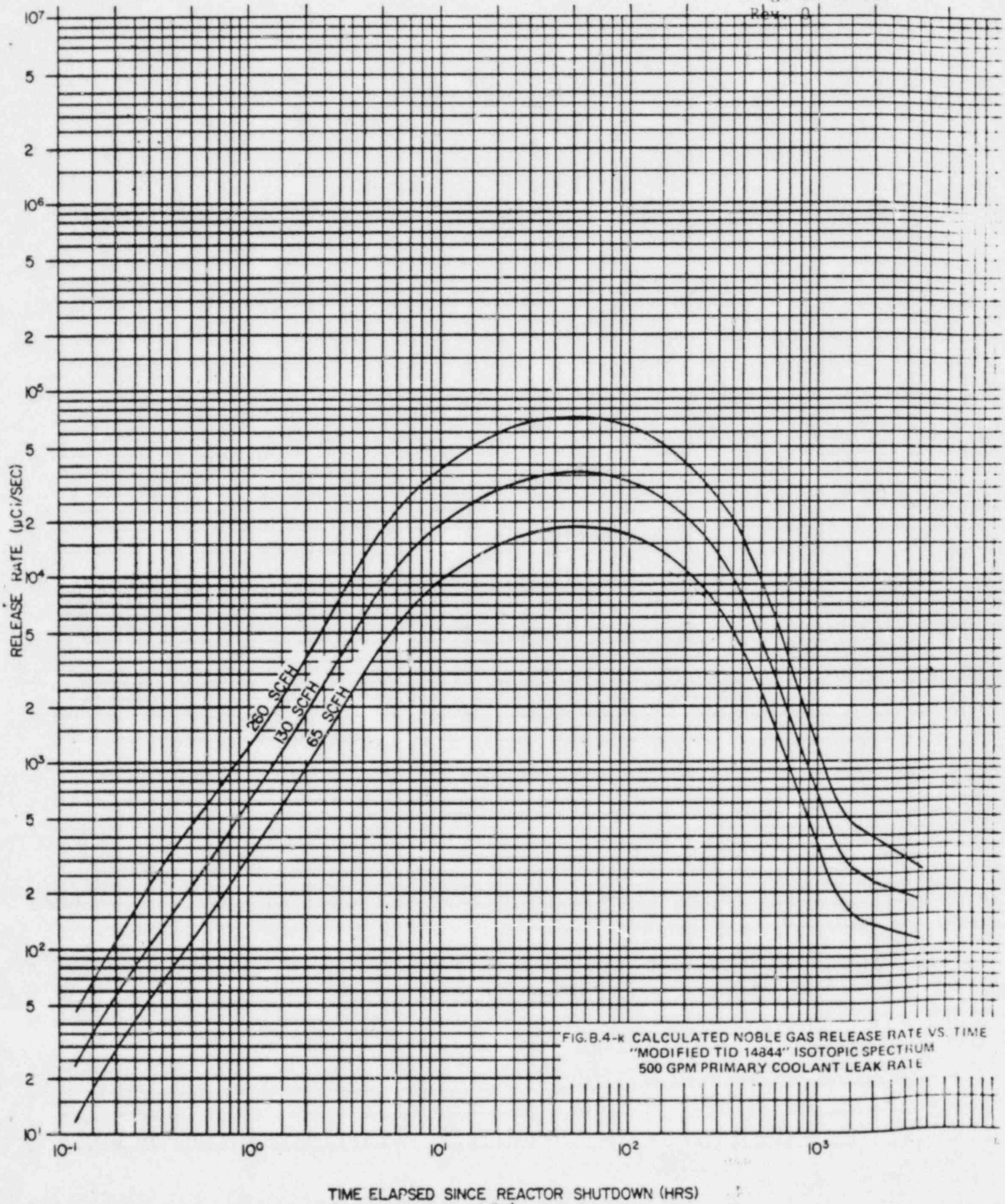


FIG. B.4-k CALCULATED NOBLE GAS RELEASE RATE VS. TIME
"MODIFIED TID 14844" ISOTOPIC SPECTRUM
500 GPM PRIMARY COOLANT LEAK RATE

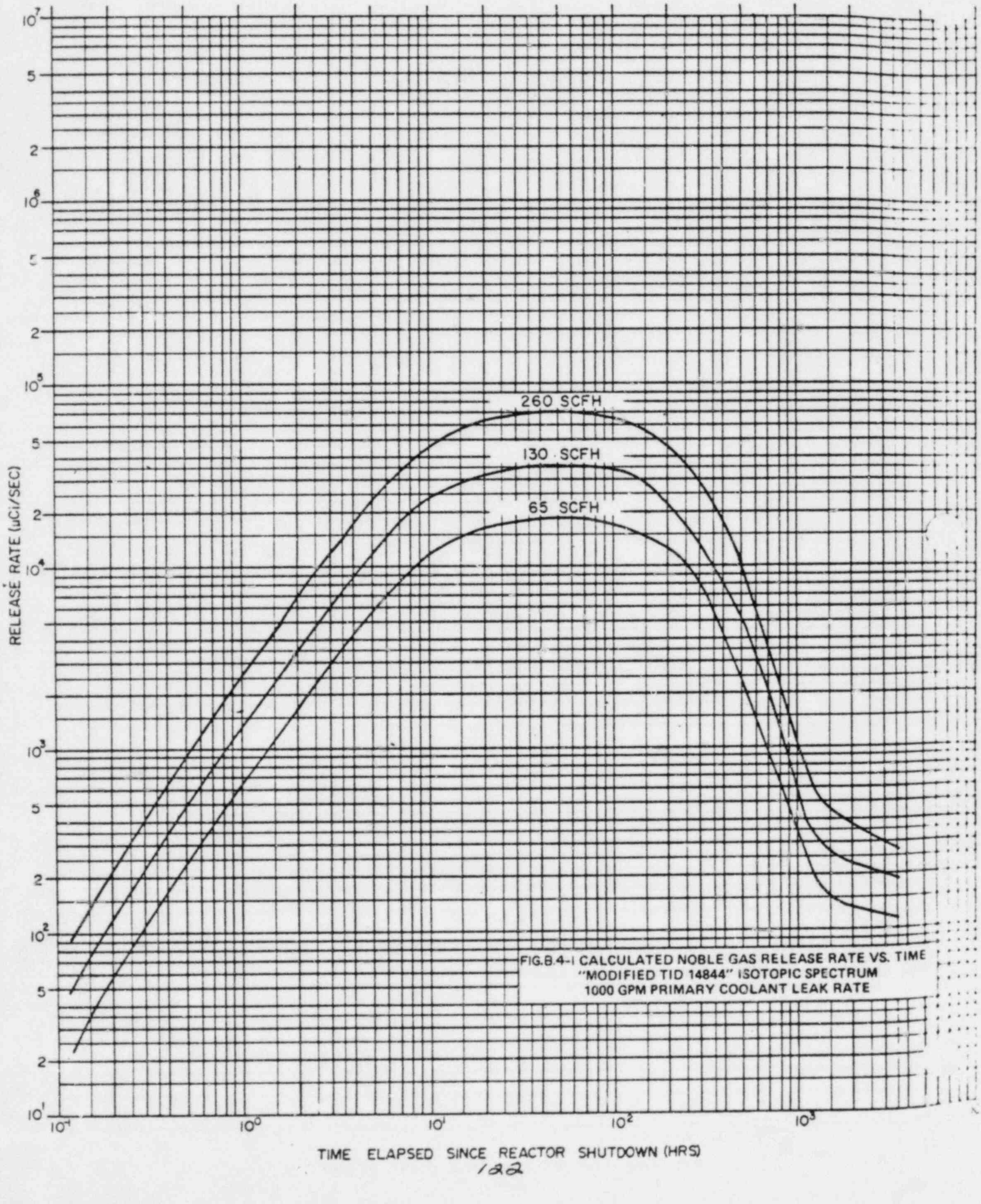


FIG.B.4-1 CALCULATED NOBLE GAS RELEASE RATE VS. TIME
"MODIFIED TID 14844" ISOTOPIC SPECTRUM
1000 GPM PRIMARY COOLANT LEAK RATE

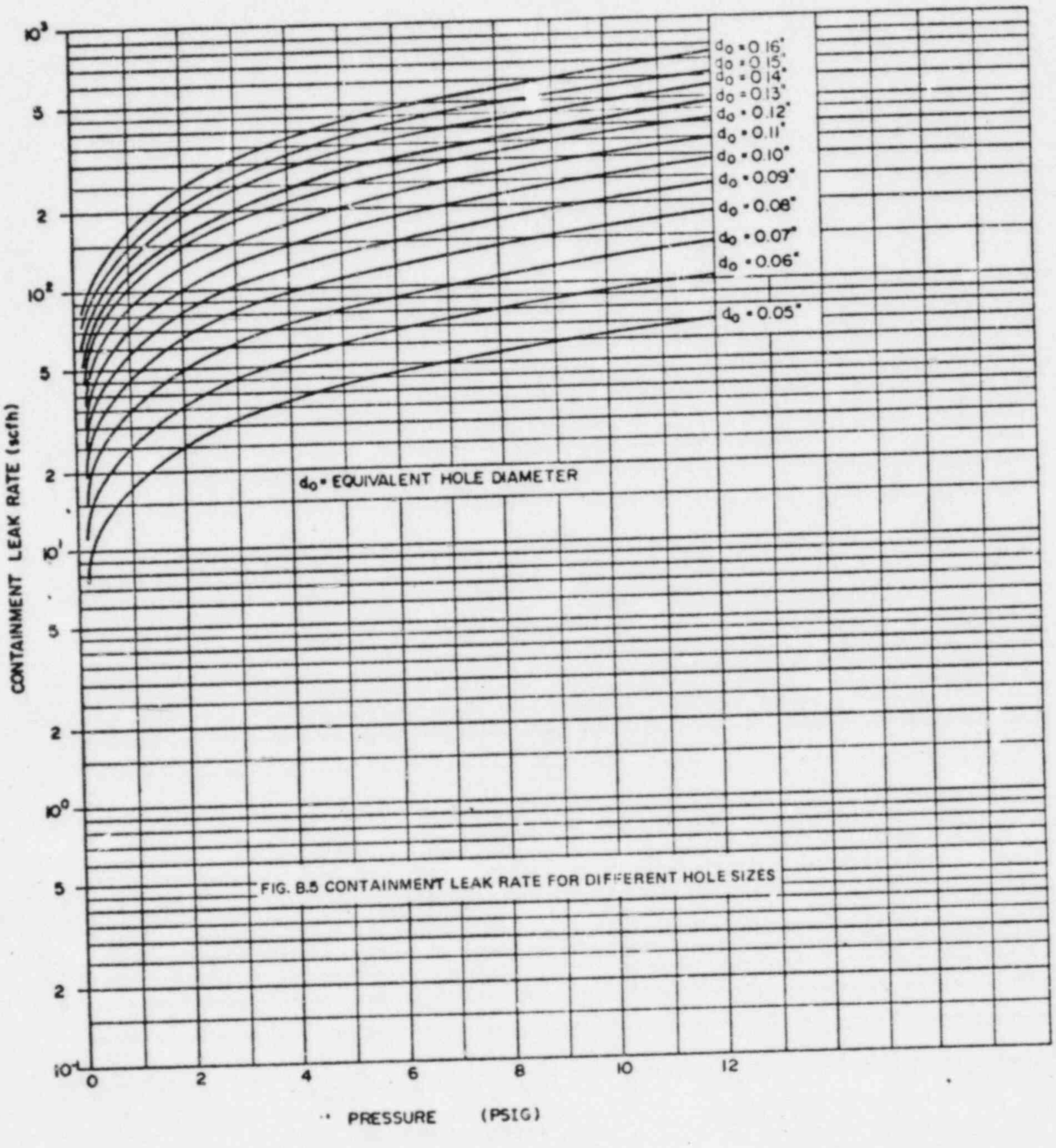


FIG. B.5 CONTAINMENT LEAK RATE FOR DIFFERENT HOLE SIZES

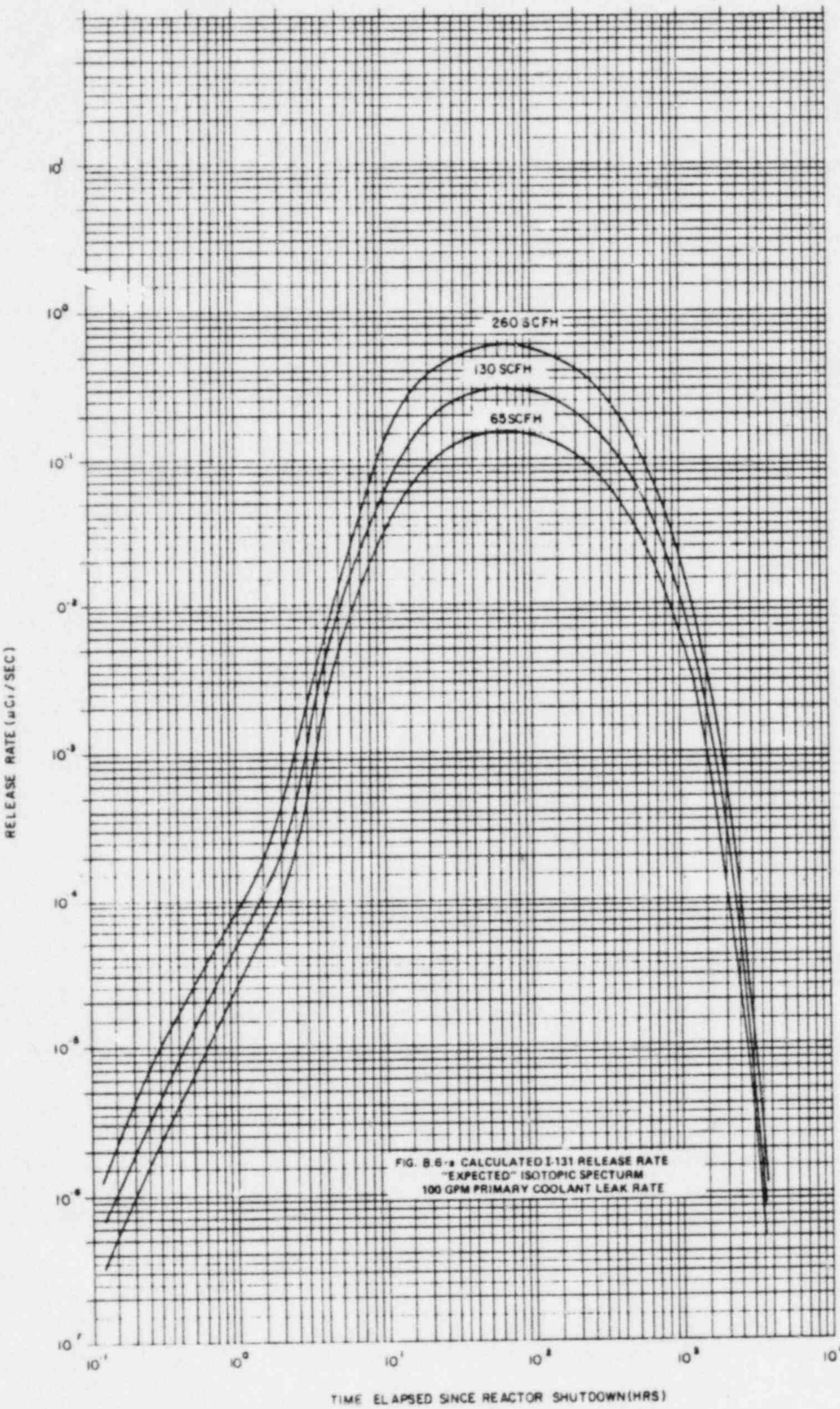
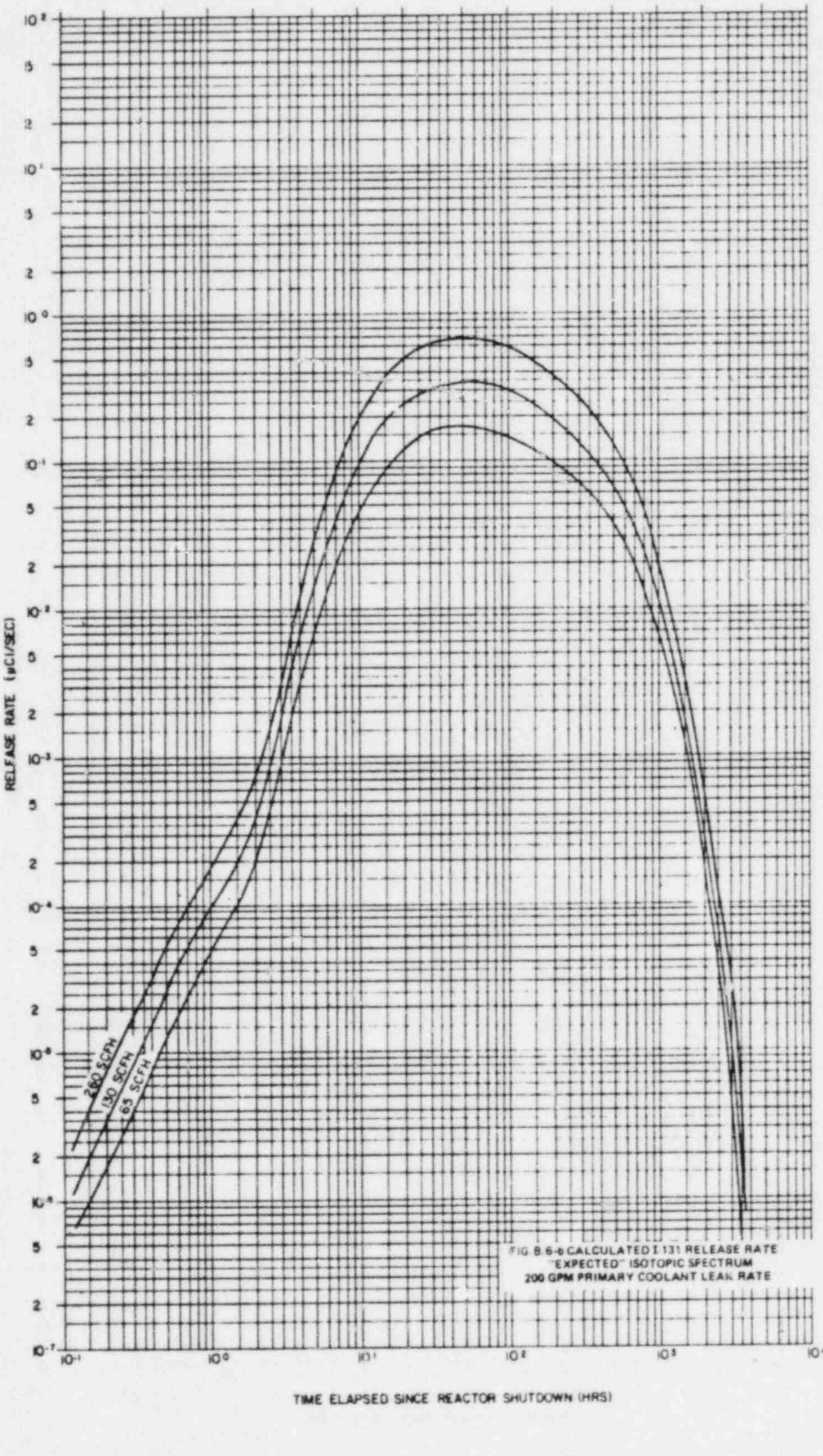


FIG. B.6-a CALCULATED I-131 RELEASE RATE
"EXPECTED" ISOTOPIC SPECTURM
100 GPM PRIMARY COOLANT LEAK RATE


RELEASE RATE (μCI/SEC)
TIME ELAPSED SINCE REACTOR SHUTDOWN (HRS)
260 SCFH
130 SCFH
65 SCFH
FIG B.6-b CALCULATED I 131 RELEASE RATE
"EXPECTED" ISOTOPIC SPECTRUM
200 GPM PRIMARY COOLANT LEAK RATE

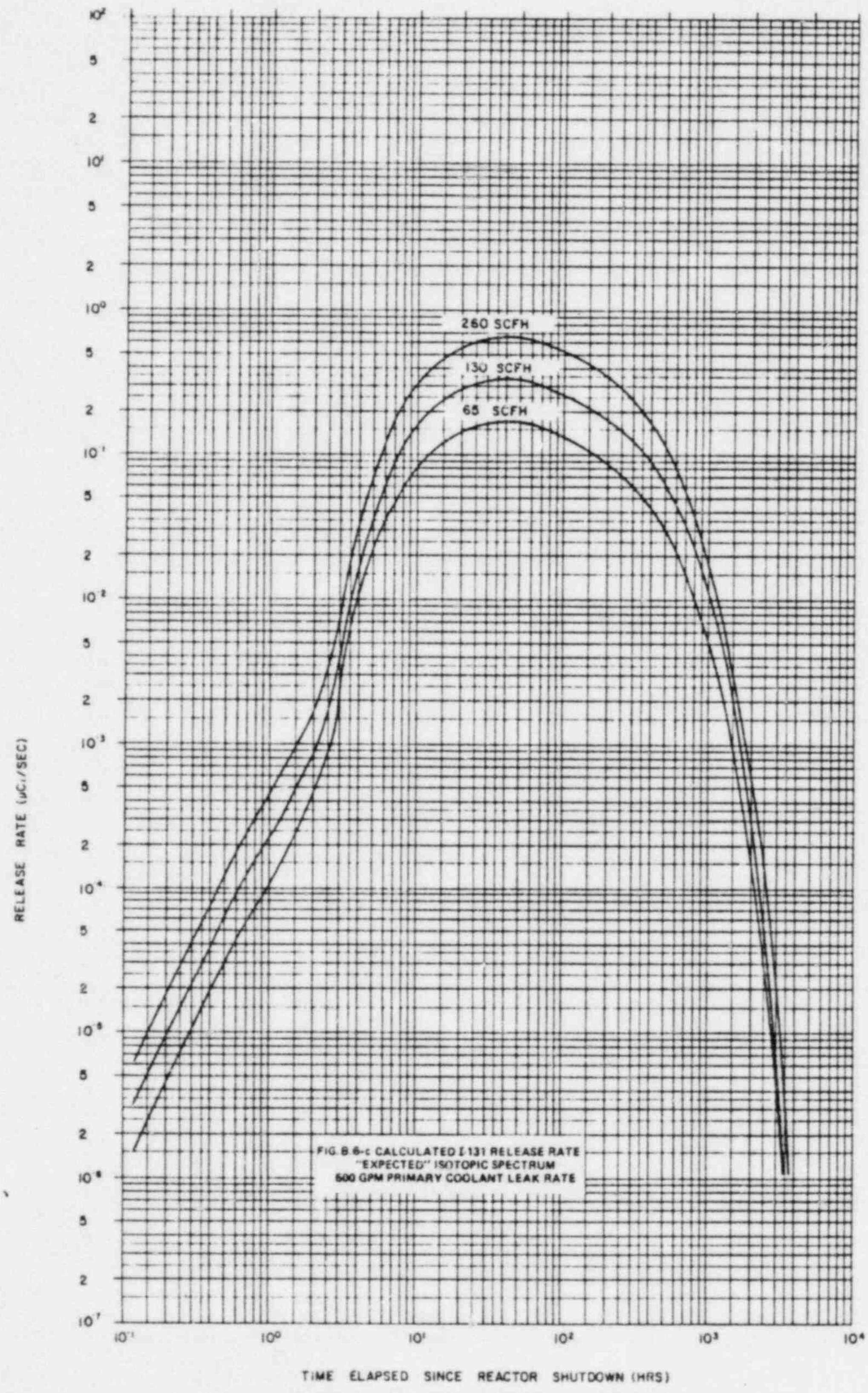
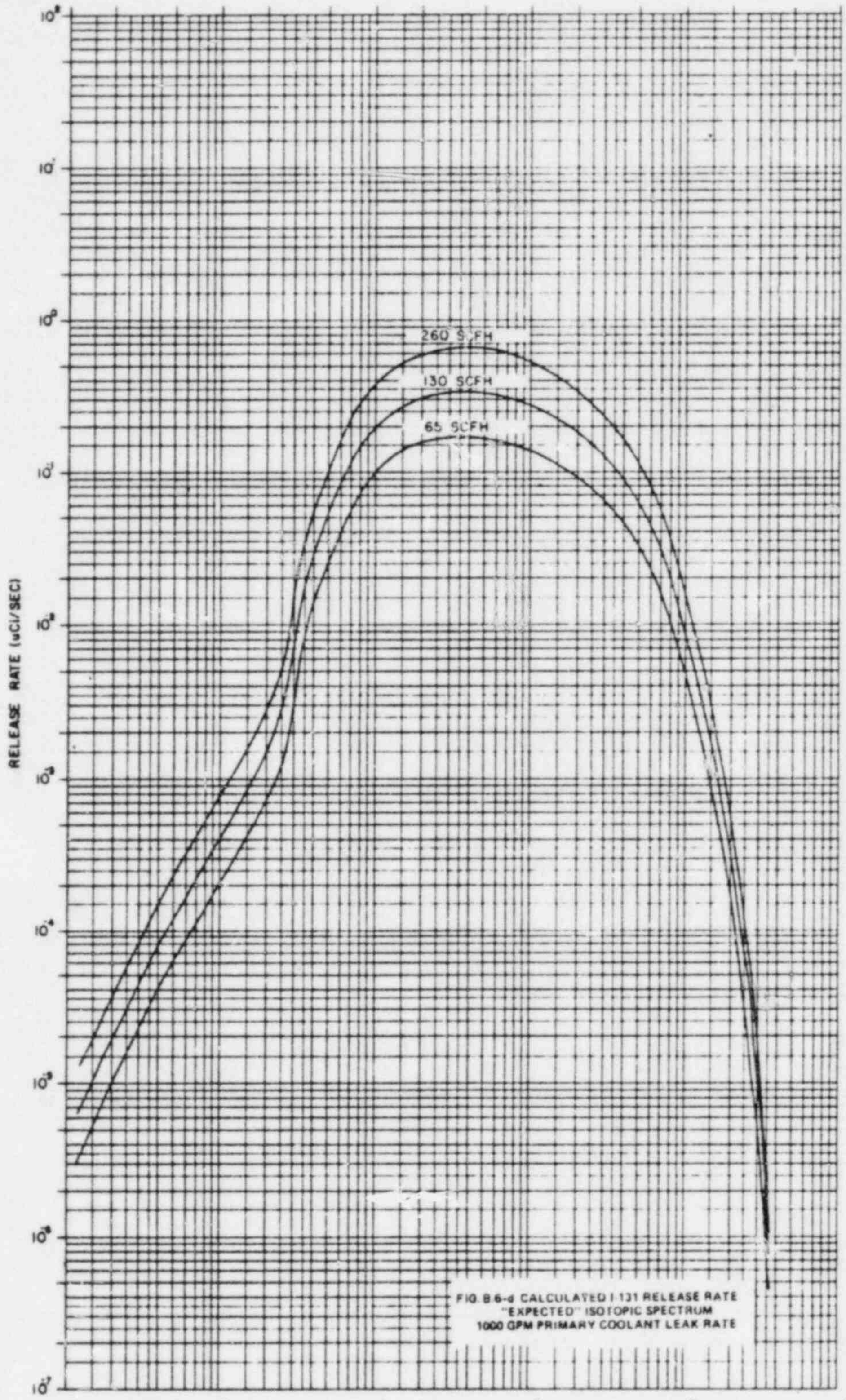


FIG. B.6-c CALCULATED I 131 RELEASE RATE
"EXPECTED" ISOTOPIC SPECTRUM
600 GPM PRIMARY COOLANT LEAK RATE


260 SCFH
130 SCFH
65 SCFH
RELEASE RATE (uCi/SEC)
TIME ELAPSED SINCE REACTOR SHUTDOWN (HRS)
FIG. B.6-d CALCULATED I-131 RELEASE RATE
"EXPECTED" ISOTOPIC SPECTRUM
1000 GPM PRIMARY COOLANT LEAK RATE

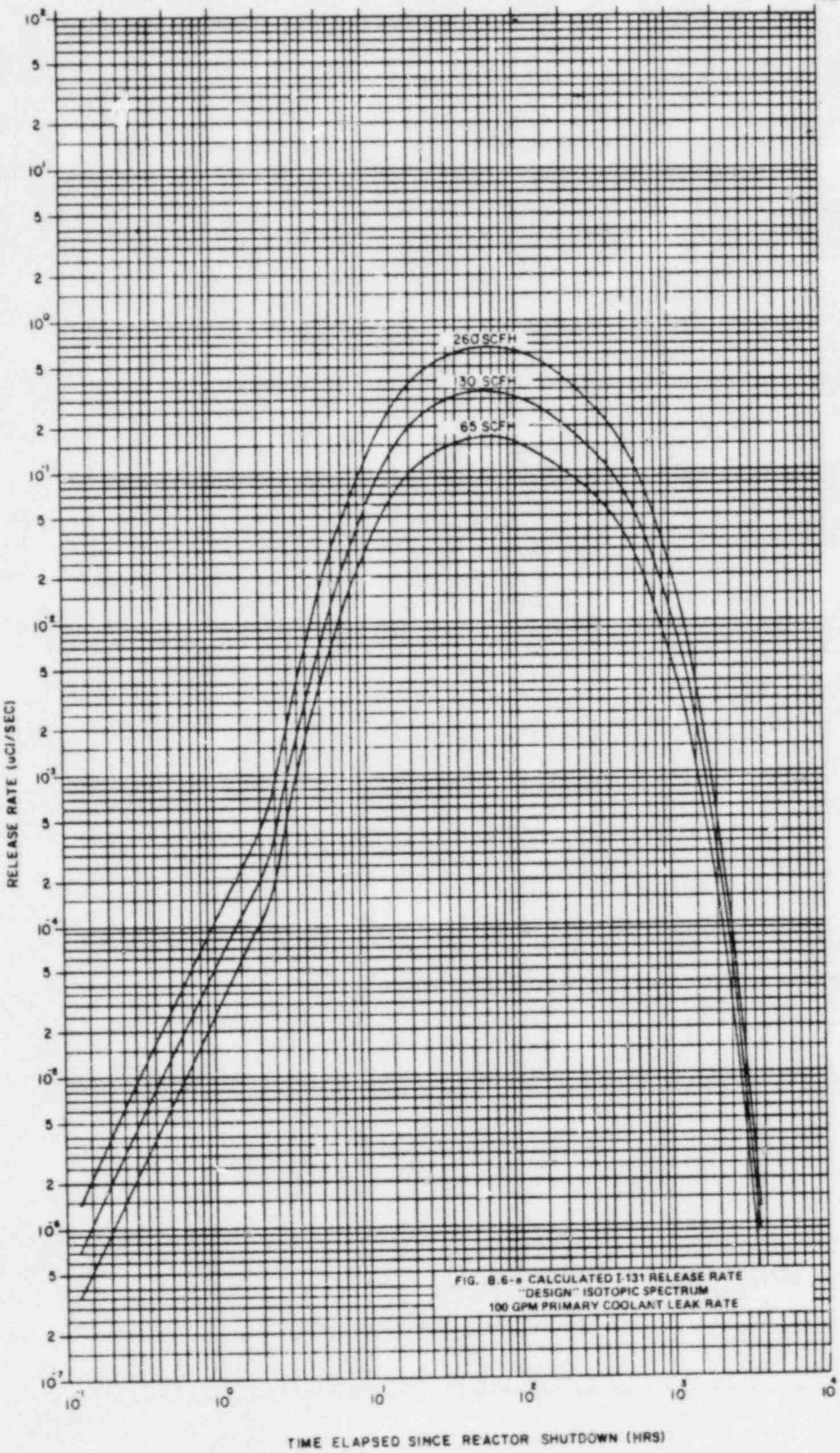


FIG. B.6-e CALCULATED I-131 RELEASE RATE
"DESIGN" ISOTOPIC SPECTRUM
100 GPM PRIMARY COOLANT LEAK RATE

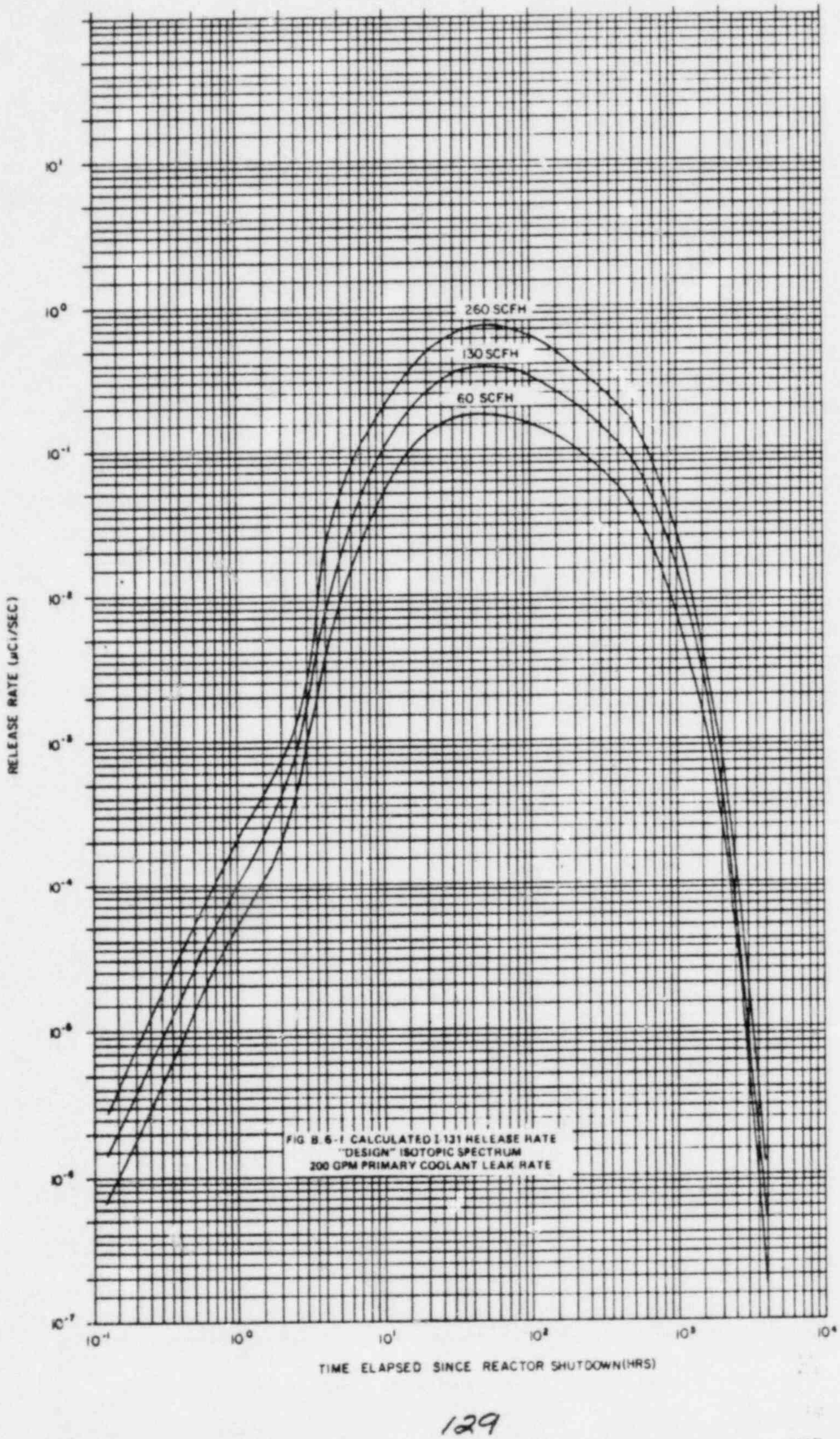


FIG B.6-f CALCULATED I 131 RELEASE RATE "DESIGN" ISOTOPIC SPECTRUM 200 GPM PRIMARY COOLANT LEAK RATE

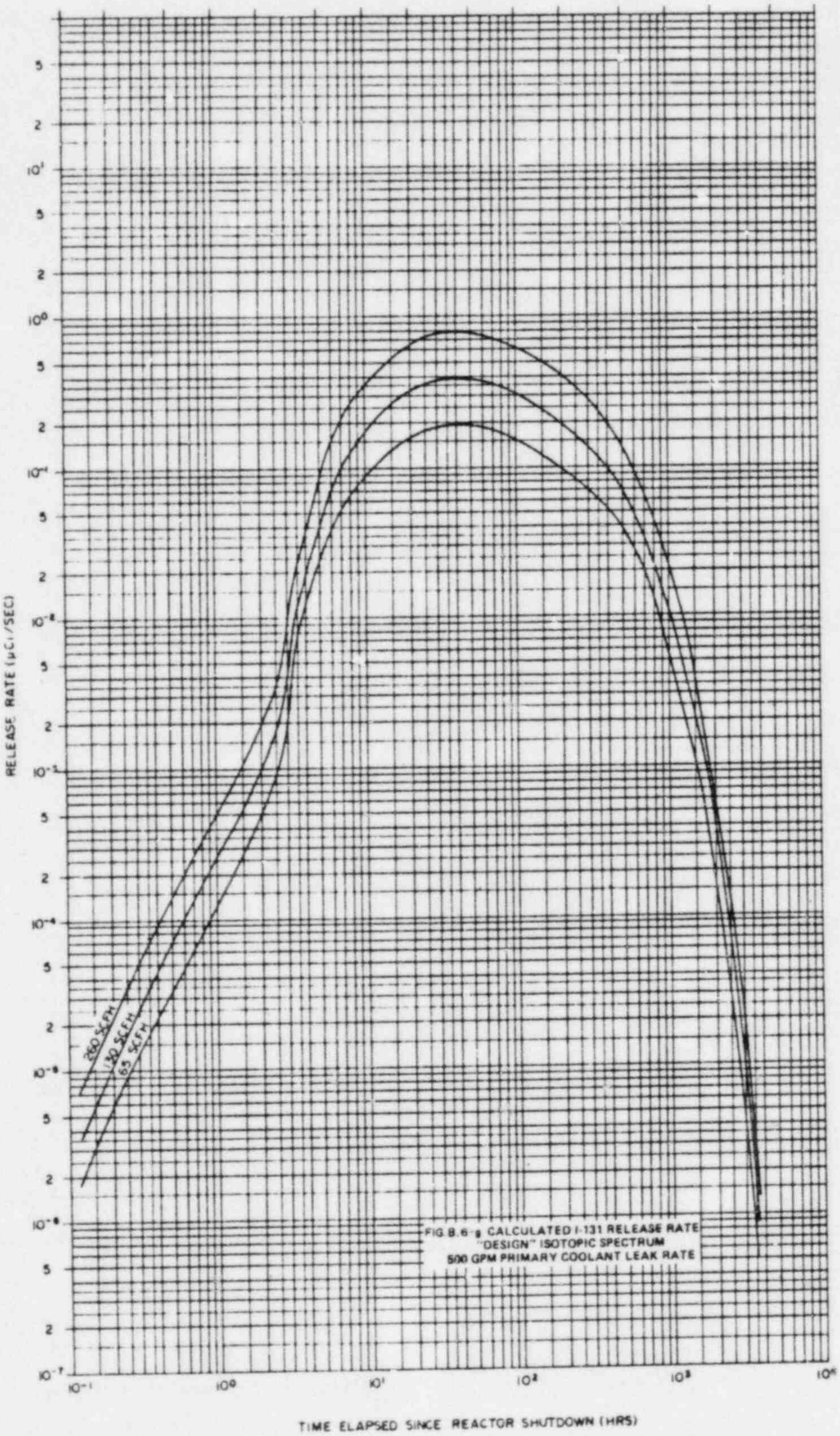
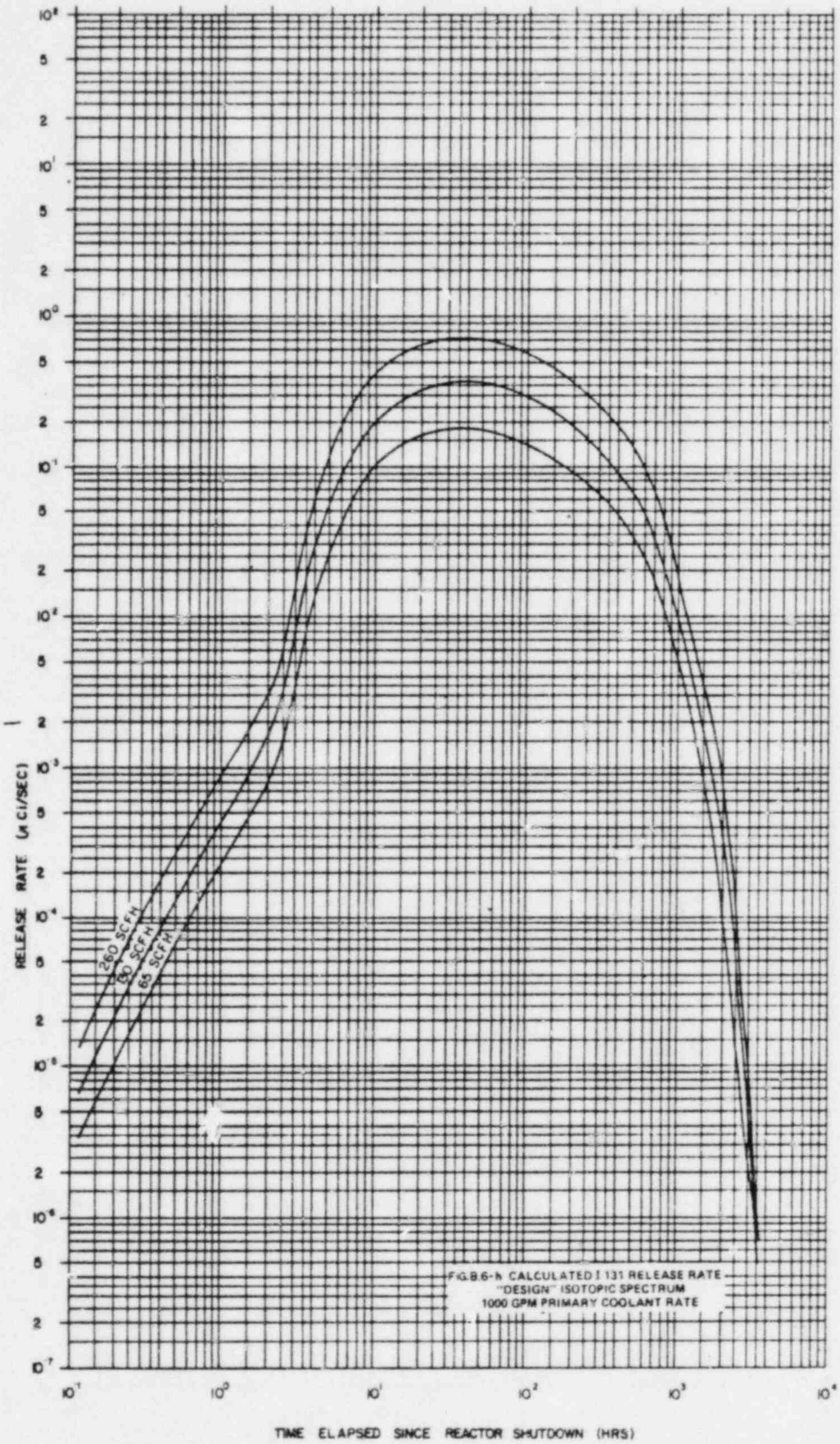


FIG. B.6-g CALCULATED I-131 RELEASE RATE
"DESIGN" ISOTOPIC SPECTRUM
500 GPM PRIMARY COOLANT LEAK RATE


RELEASE RATE (μCi/SEC)
260 SCFH
130 SCFH
65 SCFH
FIG.B.6-h CALCULATED I 131 RELEASE RATE
"DESIGN" ISOTOPIC SPECTRUM
1000 GPM PRIMARY COOLANT RATE
TIME ELAPSED SINCE REACTOR SHUTDOWN (HRS)

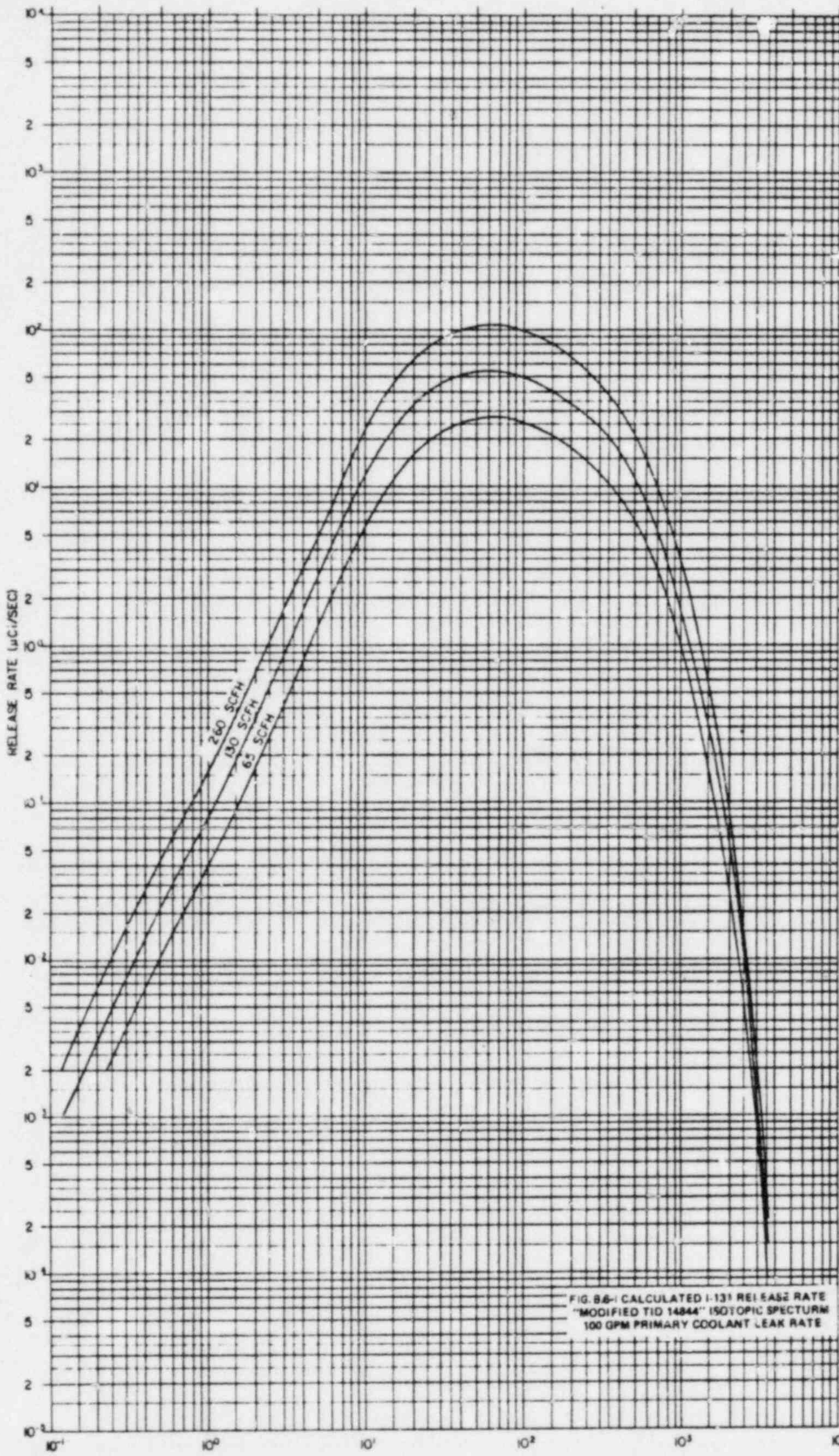


FIG. B.6-1 CALCULATED I-131 RELEASE RATE
"MODIFIED TID 14844" ISOTOPIC SPECTURM
100 GPM PRIMARY COOLANT LEAK RATE

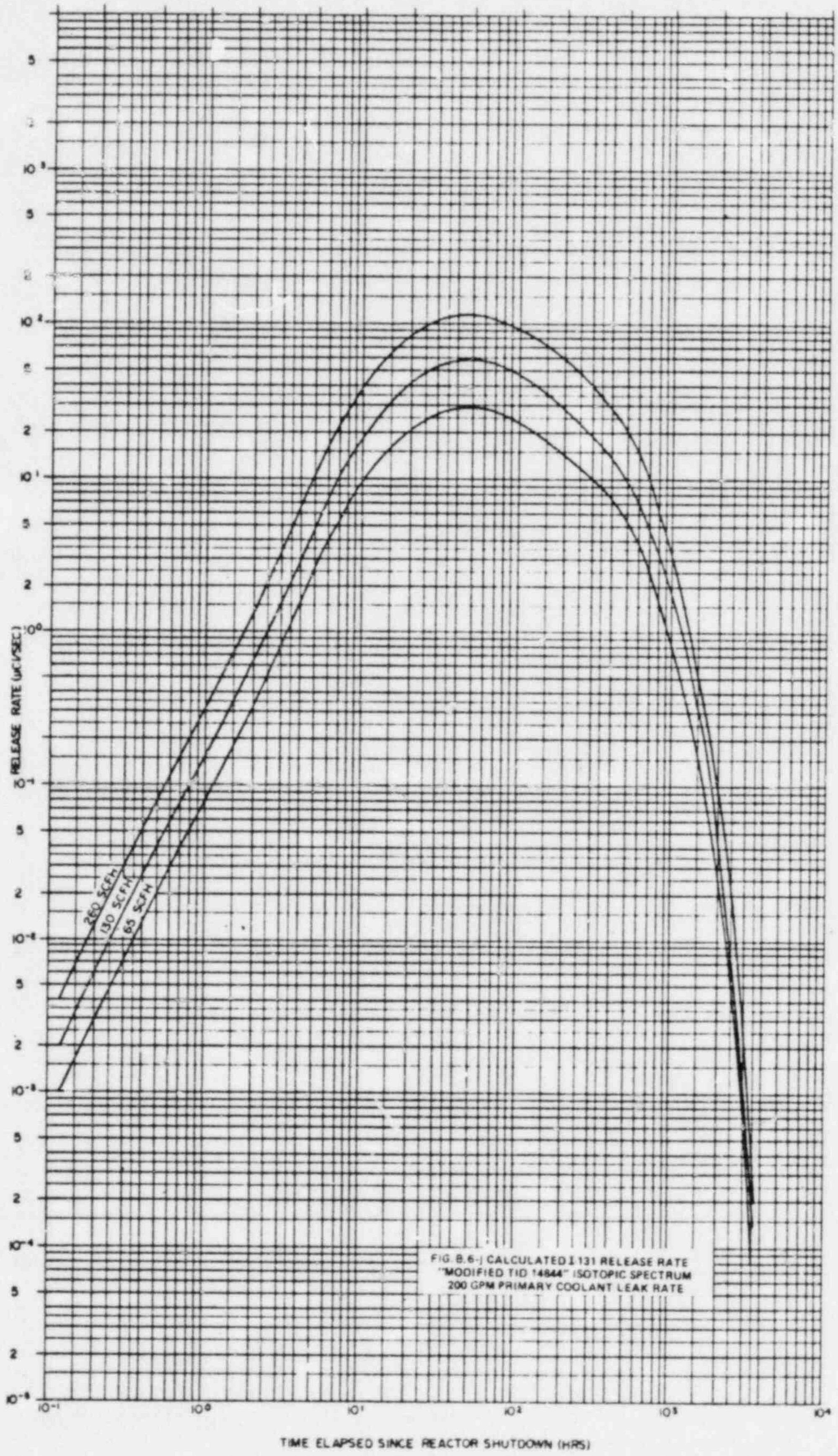


FIG. B.6-j CALCULATED I 131 RELEASE RATE
"MODIFIED TID 14844" ISOTOPIC SPECTRUM
200 GPM PRIMARY COOLANT LEAK RATE

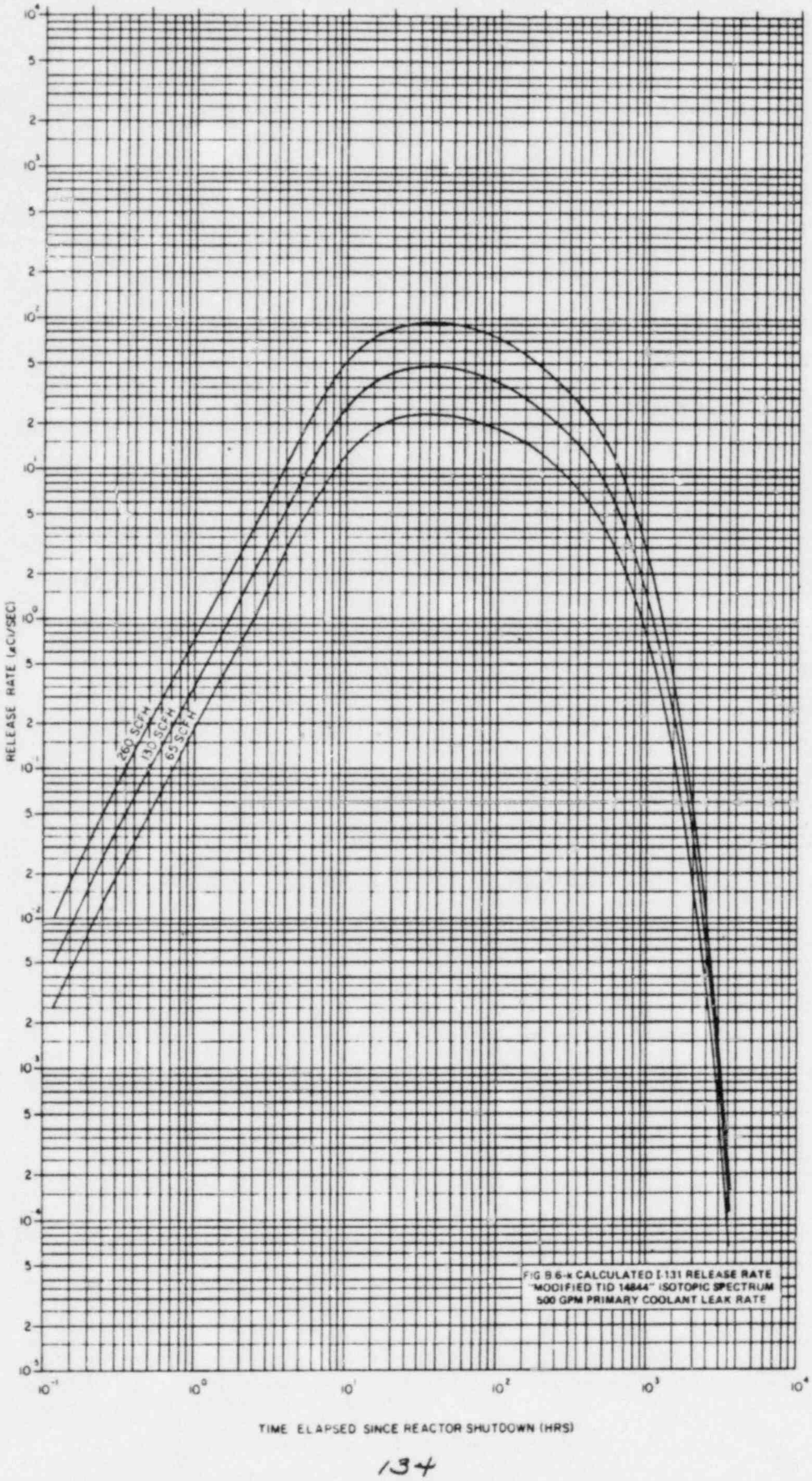


FIG B.6-k CALCULATED I-131 RELEASE RATE "MODIFIED TID 14844" ISOTOPIC SPECTRUM 500 GPM PRIMARY COOLANT LEAK RATE

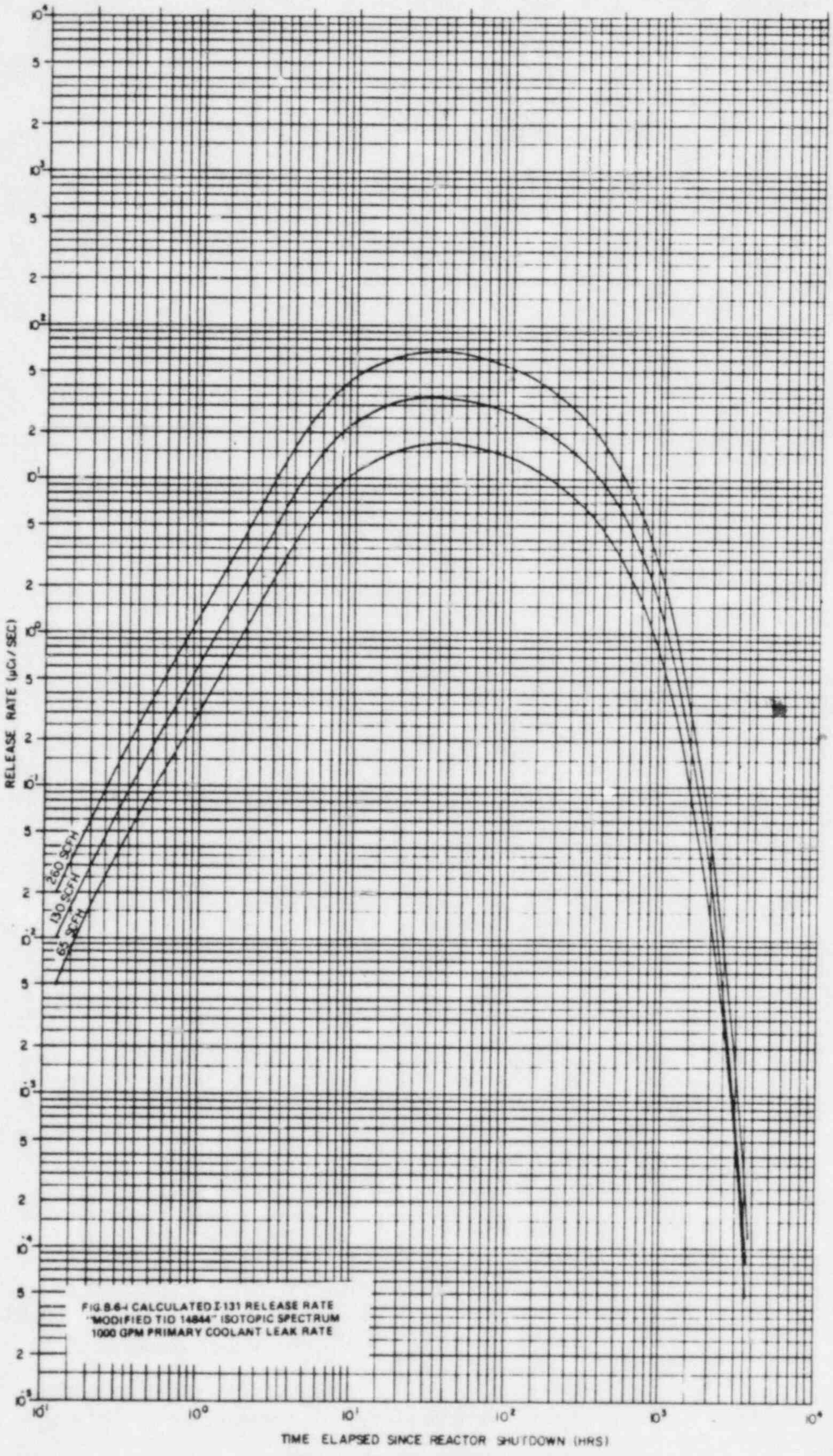


FIG. B.6-1 CALCULATED I-131 RELEASE RATE
"MODIFIED TID 14844" ISOTOPIC SPECTRUM
1000 GPM PRIMARY COOLANT LEAK RATE

1.0 PLANT PARAMETERS REQUIRED FOR RELEASE RATE CALCULATION

1.1 Determine which isotope spectrum to use based on the latest primary coolant measurement.

1.1.1 On Worksheet 1-A record the latest measured primary coolant activity (per SI-4.8 for noniodine isotopes and SI-4.9 for radioiodine, or postaccident measurement) and calculate the activity ratios.

1.1.2 On Table 1.1, circle the entries corresponding to the ratios calculated in 1.1.1.

1.1.3 On Table 1.1, circle the isotope spectrum on the line with the most circled entries. This isotope spectrum will be used in all steps of 1.0 and 2.0 of this procedure.

IMPORTANT: A POWER CHANGE OF MORE THAN 15 PERCENT WITHIN 2-8 HOURS BEFORE THE PRIMARY COOLANT SAMPLE WAS TAKEN INVALIDATES THE RATIOS INVOLVING IODINE (I). IN SUCH A CASE, RELY ON THE C/R RATIO.

1.2 Determine Normalization Factor

Note: The information labeled "Expected" and "Design" is based on 1 percent failed fuel; the "Modified TID-14844" information is based on 1 percent of the total radioactivity inventory of an equilibrium reactor core.

1.2.1 On Worksheet 1-A, circle the calculated specific activities for each of the isotopes (I-131, Cs-137, Rb-88) given in Table 1.2 for the isotope spectrum chosen in 1.1.3.

1.2.2 Calculate and record on worksheet 1-A, step 1.2.2, the ratio of the measured specific activity (from step 1.1.1) to the circled calculated values of Table 1.2.

1.2.3 Select a normalization factor corresponding most closely to two of the three ratios. Record this on worksheet 1-A, step 1.2.3.

Note: For "Expected" and "Design" isotope spectra, the normalization factor corresponds to the failed fuel fraction; for the "Modified TID-14844" spectrum, the normalization factor corresponds to the percentage of equilibrium core inventory.

1.3 Determine primary coolant leak rate into containment.

Use worksheet 1-B. Record readings of radiation monitors RM-90-59, RM-90-60 (upper compartment area monitors), and RM-90-2 (accident area monitor outside personnel hatch). Record the time and date the readings are taken. It should be noted that all radiation monitor readings are assumed to be background subtracted before being used in this predictive model. Readings taken immediately prior to the accident should be used as the background values if available. This applies to effluent as well as area monitors.

1.3.1 Calculate the elapsed time from reactor shutdown to the time at which the monitor readings were taken.

1.3.2 Record the normalization factor F, determined in 1.2.3.

1.3.3 Normalize the measured dose rates by dividing by the normalization factor. The terminology used is: MDR = measured dose rate; MDRN = normalized measured dose rate.

1.3.4.a The primary coolant leak rate will be determined using the normalized values of the upper containment area monitors (i.e., RM-90-59 and RM-90-60). Select the greater of the two normalized upper containment monitor readings determined by 1.3.3. Note that these two monitors should have similar readings. Large deviations may imply that one of the monitors is not operating properly, in which case only the reading from the operating monitor should be used. Record the normalized monitor reading which was selected as the MDRN (upper containment) in 1.3.4.a on worksheet 1-B. If the upper containment area monitors cannot be used, skip to 1.3.5.

1.3.4.b Choose the appropriate figure to be used for visual interpolation based on the reactor coolant source spectrum determined in 1.1.3:

Expected Spectrum: Figure C.1-a
Design Spectrum: Figure C.1-b
TID-14844 Spectrum: Figure C.1-c

Record the figure number selected on 1.3.4.b, worksheet 1-B.

1.3.4.c Using the elapsed time t_e from 1.3.1, and the normalized upper containment radiation (mR/hr) from 1.3.4.a, estimate a primary coolant leak rate (gpm) by visually interpolating between the curves on the figure selected in 1.3.4.b. Record in 1.3.4.c on worksheet 1-B the value of the calculated monitor reading and associated primary coolant leak rate from the curves above and below the normalized measured value.

1.3.4.d Record the estimated primary coolant leak rate in 1.3.4.d on worksheet 1-B and proceed to 1.4.

1.3.5 The primary coolant leak rate will be determined using the normalized value of the accident monitor outside the personnel hatch (i.e., RM-90-2). This monitor should be used only if the upper containment area monitors cannot be used.

1.3.5.a Choose the appropriate figure to be used for visual interpolation based on the reactor source spectrum determined in 1.1.3:

Expected Spectrum: Figure C.2-a
Design Spectrum: Figure C.2-b
TID-14844 Spectrum: Figure C.2-c

Record the figure number selected on 1.3.5.a, worksheet 1-B.

1.3.5.b Using the elapsed time t_e from 1.3.1 and the normalized radiation level (mR/hr) outside the personnel hatch from 1.3.3, estimate a primary coolant leak rate (gpm by visually interpolating between the curves on the figure selected in 1.3.5.a. Record in 1.3.5.b on worksheet 1-B the value of the calculated monitor reading and associated primary coolant leak rate from the curves above and below the normalized measured value.

1.3.5.c Record the estimated primary coolant leak rate in 1.3.5.c on worksheet 1-B. Proceed to 1.4.

1.4 <u>Determine containment leak rate.</u>

Use worksheet 1-C. Record the readings of radiation monitor RM-90-100B (shield building vent - gross radioactive gas channel) and flow element FE-30-242 (shield building vent total flow). Record the time and date the readings were taken.

Note: If the readings of the shield building vent monitor are not significantly greater than the normal background count rate, the containment leak rate cannot be determined analytically. For this situation, the containment leak rate will be assumed to be 268 scfh (0.25 percent per day of primary containment free volume at 12.0 psig). If this assumption is made, skip to 2.0.

1.4.1 Calculate the noble gas specific activity (Ci/cc) in the shield building vent by multiplying the monitor reading (cpm) by the conversion factor of 2.8×10^{-8} Ci/cc/cpm:

Vent Concentration = Monitor cpm x 2.8×10^{-8}

1.4.2 Calculate the noble gas release rate (μCi/s) by multiplying the specific activity (μCi/cc) determined in 1.4.1 by the vent flow (cfm) from FE-30-242 and by 472 (unit conversion factor):

Release rate = 472 x specific activity x vent flow

1.4.3 Calculate the elapsed time from reactor shutdown to the time at which the shield building vent radiation monitor reading was taken.

1.4.4 Record the normalization factor F, determined in 1.2.3.

1.4.5 Normalize the measured noble gas release rate by dividing by the normalization factor. The terminology used is: VRRN = normalized vent noble gas release rate.

1.4.6 Select from figure C.4, the figure to be used for visual interpolation based on the reactor coolant source spectrum determined in 1.1.3 and the primary coolant leak rate determined in 1.3.4.d or 1.3.5.c. Record the figure number selected on 1.4.6, worksheet 1-C.

1.4.7 Using the elapsed time from 1.4.2 and the normalized noble gas release rate determined in 1.4.5, estimate a primary containment leak rate (scfh) by visually interpolating between the curves on the figure selected in 1.4.6. Record in 1.4.7 on worksheet 1-C the value of the calculated noble gas release rate and associated primary containment leak rate from the curves above and below the normalized measured value.

1.4.8 Record the estimated primary containment leak rate in 1.4.8 on worksheet 1-C.

1.4.9 Record the readings of primary containment pressure monitors PdI-30-42, PdI-30-43, PdI-30-44, and PdI-30-45 on worksheet 1-C. Also, record the time at which the readings were taken.

1.4.10 From figure C.4, obtain an equivalent hole size corresponding to the estimated primary containment leak rate from step 1.4.8 and the containment pressure. Record the hole size diameter (inches) in 1.4.10 on worksheet 1-C.

2.0 PREDICTED VENT RELEASE RATE--NO CHANGE IN PLANT CONDITIONS

2.1 Calculate noble gas vent release rate

Use worksheet 2-A.

2.1.1 Use the same sheet of Figure C.3 selected in Step 1.4.6.

2.1.2 Use the containment leak rate determined in Step 1.4.8.

2.1.3 Select the time for which the vent release rate is desired.

2.1.4 Obtain the noble gas vent release rate by interpolation between the curves on Figure C.3:

2.1.4.a Determine interpolation fraction = $\frac{LRD - LR(L)}{LR(U) - LR(L)}$

where: LRD = containment leak rate from Step 1.4.8.
LR = containment leak rate for which curve is calculated.
(U),(L) = value of LR immediately (above) (below) LRD

2.1.4.b Determine the difference between vent release rates corresponding to LR(U) and LR(L) at the desired time, multiply by the interpolation fraction, and add to the vent release rate corresponding to LR(L).

2.1.4.c Multiply by the normalization factor from Step 1.2.2.

2.2 Calculate the iodine vent release rate.

Use worksheet 2-B.

2.2.1 Use the sheet of Figure C.5 corresponding to the one of Figure C.3 selected in Step 2.1.1.

2.2.2 Follow procedure 2.1. Substitute Figure C.5 for Figure C.3.

NOTE: It is not necessary to recalculate the interpolation fraction (Step 2.1.4a) since it will be unchanged.

3.0 <u>PREDICTED VENT RELEASE RATE--CHANGING PLANT CONDITIONS</u>

3.1 <u>For a different postulated primary coolant activity:</u>

3.1.1 Consider which isotope spectrum ("Expected," "Design," "Modified TID-14844") was chosen in Step 1.1.

3.1.2 Consider the magnitude of the normalization factor determined in Step 1.2.

3.1.3 Decide on a new primary coolant activity by increasing the normalization factor (fuel failure) or changing the isotope spectrum based on plausible developments in the condition of the plant. In order of increasing severity, isotope spectra rank as follows:

(1) Expected

(2) Design

(3) Modified TID-14844

If the isotope spectrum is "Expected," do not increase the normalization factor to more than 2; instead, change the spectrum to "Design."

If the isotope spectrum is "Design," do not increase the normalization factor to more than 15; instead, change the spectrum to "Modified TID-14844."

3.2 <u>For a different postulated containment leak rate</u>

3.2.1 For an anticipated change in containment pressure with no additional containment degradation, use Figure C.4 with the equivalent hole size determined in Step 1.4.10.

3.2.2 For anticipated containment degradation, use Figure C.3 with an appropriately chosen hole size.

3.2.3 Determine the new containment leak rate from Figure C.3.

3.3 <u>For an anticipated degradation of the primary coolant loop.</u>

Change the primary coolant leak rate to containment.

3.4 For the postulated changed plant parameters, recalculate future vent releases using the methods given in part 2.0.

NOTE: Part 2.0 is based on unchanged plant parameters. If the postulated conditions are more severe than those determined to exist in part 1.0, the part 2.0 methods with the changed parameters will result in release rates which are too high; conversely, in the unlikely event that the postulated conditions are less severe, the estimated release rates will be too low.

GASEOUS RELEASES WORKSHEET
1-A

1.1 Determination of Isotopic spectrum

1.1.1 I_m = I-131 specific activity ________ μ Ci/g
C_m = Cs-137 specific activity ________ μ Ci/g
R_m = Rb-88 specific activity ________ μ Ci/g

I_m/C_m = ________ I_m/R_m = ________ C_m/R_m = ________

1.1.2

TABLE 1.1
Selection of Isotope Spectrum

I/C	I/R	C/R	Appropriate Isotope Spectrum
8.5 I/C 18.5	1.0 I/R 10	C/R 0.18	Expected
I/C 8.5	I/R 1.0	0.18 C/R 1.0	Design
I/C 18.5	I/R 10	C/R 1.0	Modified TID-14844

1.1.3 Chosen distribution is (check one):

___ "Expected"
___ "Design"
___ "Modified TID-14844"

1.2 Determination of severity of fuel failure:

1.2.1

Table 1.2
Specific Activity in Primary Coolant (Prior to Incident) for Three Isotopic Spectra

Isotope	Specific Activity Level (μCi/g) Expected	Design	Modified TID-14844
I_c = I-131	2.3	2.5	3.57×10^3
C_c = Cs-137	0.16	1.0	1.59×10^2
R_c = Rb-88	1.9	3.7	9.09×10^1

GASEOUS RELEASES WORKSHEET
1-A (Continued)

1.2.2 Subscript m denotes measured specific activity
Subscript c denotes calculated specific activity from Table 2.

I_m/I_c = ________ C_m/C_c = ________ R_m/R_c = ________

1.2.3 Select a normalization factor corresponding most closely to two of the three ratios:

Normalization Factor F = ____

GASEOUS RELEASES WORKSHEET
1-B

1.3 Radiation Monitor Readings ______________________
Time and Date

RM-90-59: _______ mR/hr
RM-90-60: _______ mR/hr
RM-90-2 : _______ mR/hr

1.3.1 Reactor shutdown date and time _______________
Monitor reading date and time _______________
Elapsed time t_e = _______ hr _____ min = _____ hrs

1.3.2 Normalization Factor (from 1.2.3, worksheet 1-A) F = _______

1.3.3 MDRN (RM-90-59) = MDR/F = ________ / ________ = ______ mR/hr
(RM-90-59) F

MDRN (RM-90-60) = MDR/F = ________ / ________ = ______ mR/hr
(RM-90-60) F

MDRN (RM-90-2) = MDR/F = ________ / ________ = ______ mR/hr
(RM-90-2) F

1.3.4.a MDRN (Upper Containment) = __________ mR/hr

1.3.4.b Figure used for visual interpolation: Figure C.1 _____

1.3.4.c CCDR(U) = __________ mR/hr; Upper Leak Rate = __________ gpm

CCDR(L) = __________ mR/hr; Lower Leak Rate = __________ gpm

where: CCDR(U) = calculated upper containment monitor reading greater than MDRN (upper containment).
CCDR(L) = calculated upper containment monitor reading lower than MDRN (upper containment).

1.3.4.d Estimated primary coolant leak rate: __________ gpm

GASEOUS RELEASES WORKSHEET
1-B (Continued)

1.3.5.a Figure used for visual interpolation: Figure C.2 ______

1.3.5.b CPDR(U) = __________ mR/hr; Upper Leak Rate = __________ gpm

CPDR(L) = __________ mR/hr; Lower Leak Rate = __________ gpm

where: CPDR(U) = calculated personnel hatch monitor reading greater than MDRN (personnel hatch).
CPDR(L) = calculated personnel hatch monitor reading lower than MDRN (personnel hatch).

1.3.5.c Estimated primary coolant leak rate: __________ gpm

GASEOUS RELEASES WORKSHEET
1-C

1.4 RM-90-100B: ________ cpm at ____________ Time and Date

FE-30-242: ________ cfm at ____________ Time and Date

1.4.1 Measured shield building vent noble gas specific activity (MSA):

$$MSA = \frac{\_\_\_\_\_\_}{\text{RM-90-100B}} \text{ cpm} \times (2.8 \times 10^{-8} \mu Ci/cc/cpm) = \_\_\_\_\_\_ \mu Ci/cc$$

1.4.2 Measured shield building noble gas release rate (VRR):

$$VRR = 472 \times \frac{\_\_\_\_\_\_}{MSA} \times \frac{\_\_\_\_\_\_}{\text{FE-30-242}} = \_\_\_\_\_\_ \mu Ci/s$$

1.4.3 Reactor shutdown time and date ____________
RM-90-100B reading time and date ____________
Elapsed time t_e = ______ hr ______ min = ______

1.4.4 Normalization factor (from 1.2.3 worksheet 1-A) F = ________

1.4.5 VRNN = VRR/F = ______ / ______ = ______ μCi/s

1.4.6 Figure used for visual interpolation: Figure C.3.____

1.4.7 VRRC(U) = _______ μCi/s at containment leak rate of _______ scfh

VRRC(L) = _______ μCi/s at containment leak rate of _______ scfh

1.4.8 Containment gaseous leak rate LRD:

LRD = ________ scfh

1.4.9 Containment pressure readings ____________ Time and date

Containment Pressure PdI-30-42: ______ psig
PdI-30-43: ______ psig
PdI-30-44: ______ psig
PdI-30-45: ______ psig

1.4.10 Equivalent hole size diameter: ________ in.

GASEOUS RELEASES WORKSHEET
2-A
Containment Noble Gas Release Rate

2.1.2 From Figure C.3 ___ (Record figure used)

LR(U) = _____ scfh   LR(L) = _____ scfh

where: LR = Containment leak rate used in Step 1.4.4.

Containment Leak Rate from Worksheet 1-C, step 1.4.5:

LRD = _____ scfh

2.1.3 Time elapsed since reactor shutdown t: = _____ hours

2.1.4 Interpolation Fraction:

$$IF = \frac{LRD - LR(L)}{LR(U) - LR(L)}$$

$$= \frac{(\quad\quad) - (\quad\quad)}{(\quad\quad) - (\quad\quad)} = \_\_\_\_\_\_\_\_$$

From Figure C.3, at t, determine CRR(U) corresponding to LR(U):

CRR(U) = _____ μCi/s

From Figure C.3, at t, determine CRR(L) corresponding to LR(L):

CRR(L) = _____ μCi/s

Normalization factor (from 1.2.3, worksheet 1-A) F = _____

Future noble gas release rate =

[IF x (CRR(U) - CRR(L)) + CRR(L)] x F

= [_____ x (_____ - _____) + _____] x _____

= _____ μCi/s

NOTE: Same nomenclature as on worksheet 1-C.

GASEOUS RELEASES WORKSHEET
2-B

Interpolation Fraction from Worksheet 2-A: IF = ____

2.2.1 From Figure C.5 __ : (Same nomenclature as worksheet 2-A).

CRR(L) = ____ μCi/s   CRR(U) = ____ μCi/s

Normalization Factor F = ____

Future iodine release rate:

[IF x (CRR(U) - CRR(L)) + CRR(L)] x F

= [____ x (____ - ____) + ____] x ____

= ____ μCi/s

FIGURES

FIGURES C.1

FIGURES C.2

FIGURES C-3

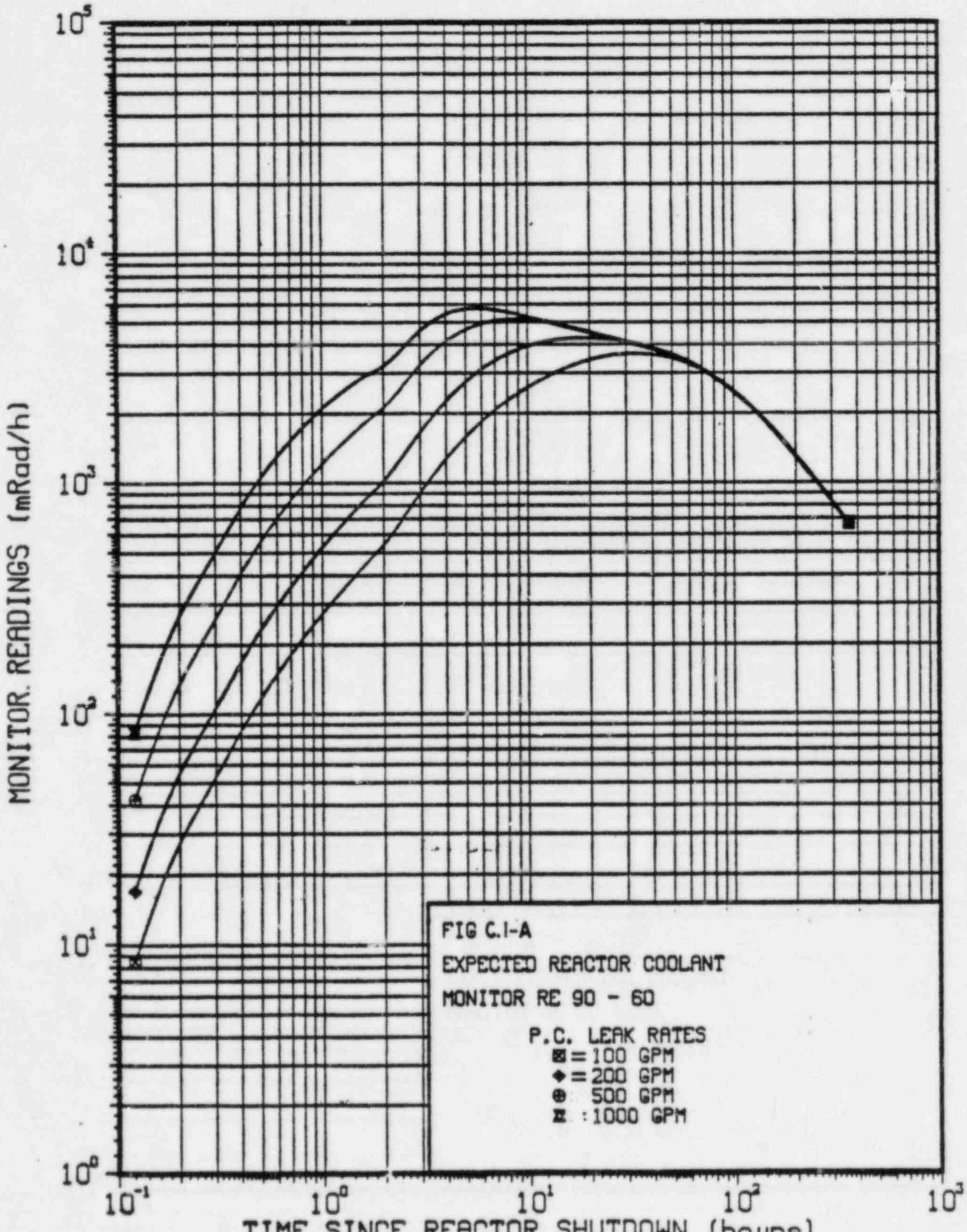
FIGURE C.4

C.4 Containment leak rate versus containment pressure and hole size.

FIGURES C.5

CONTAINMENT MONITOR READINGS vs. TIME

FOR DIFFERENT PRIMARY COOLANT LEAK RATES



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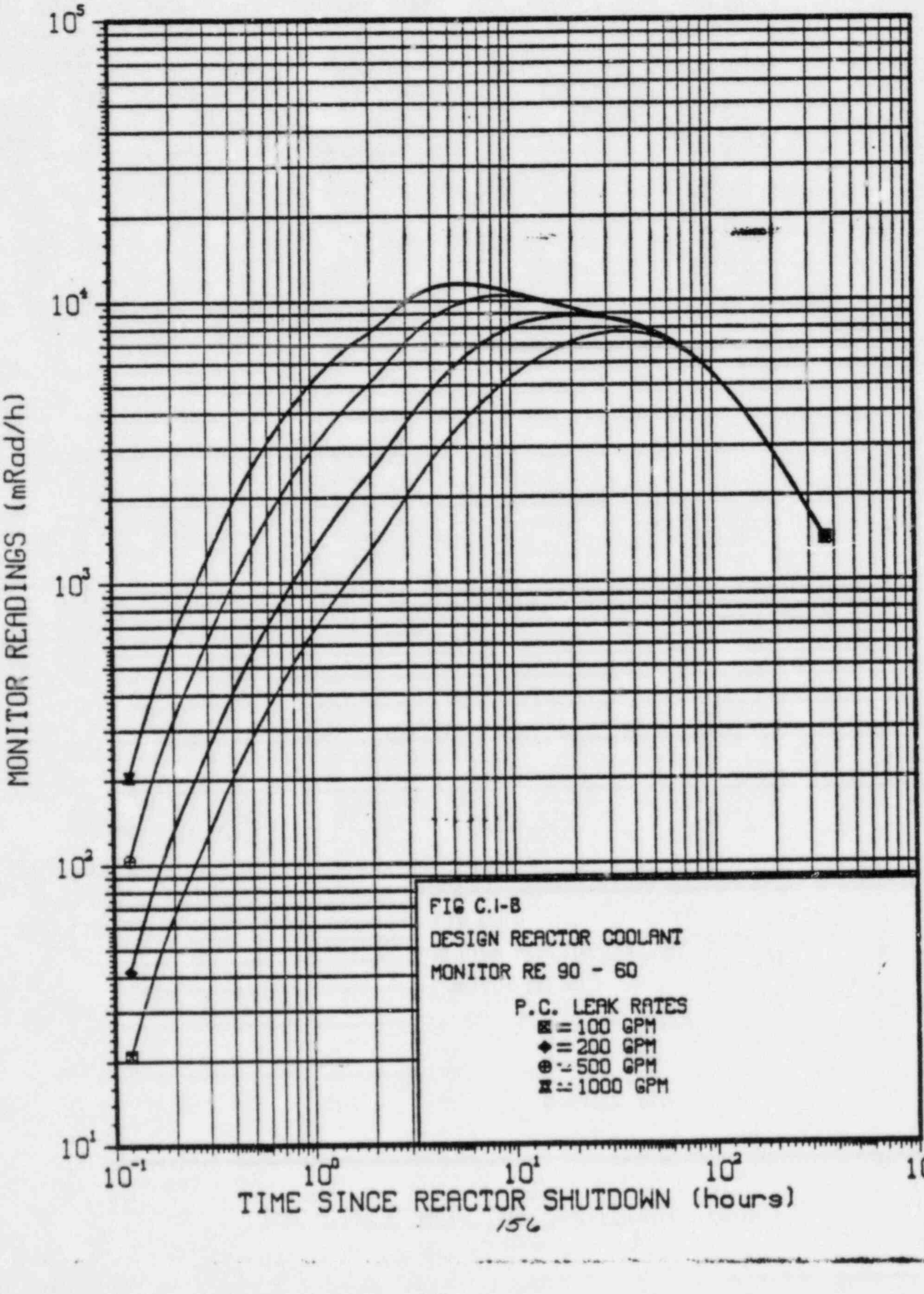
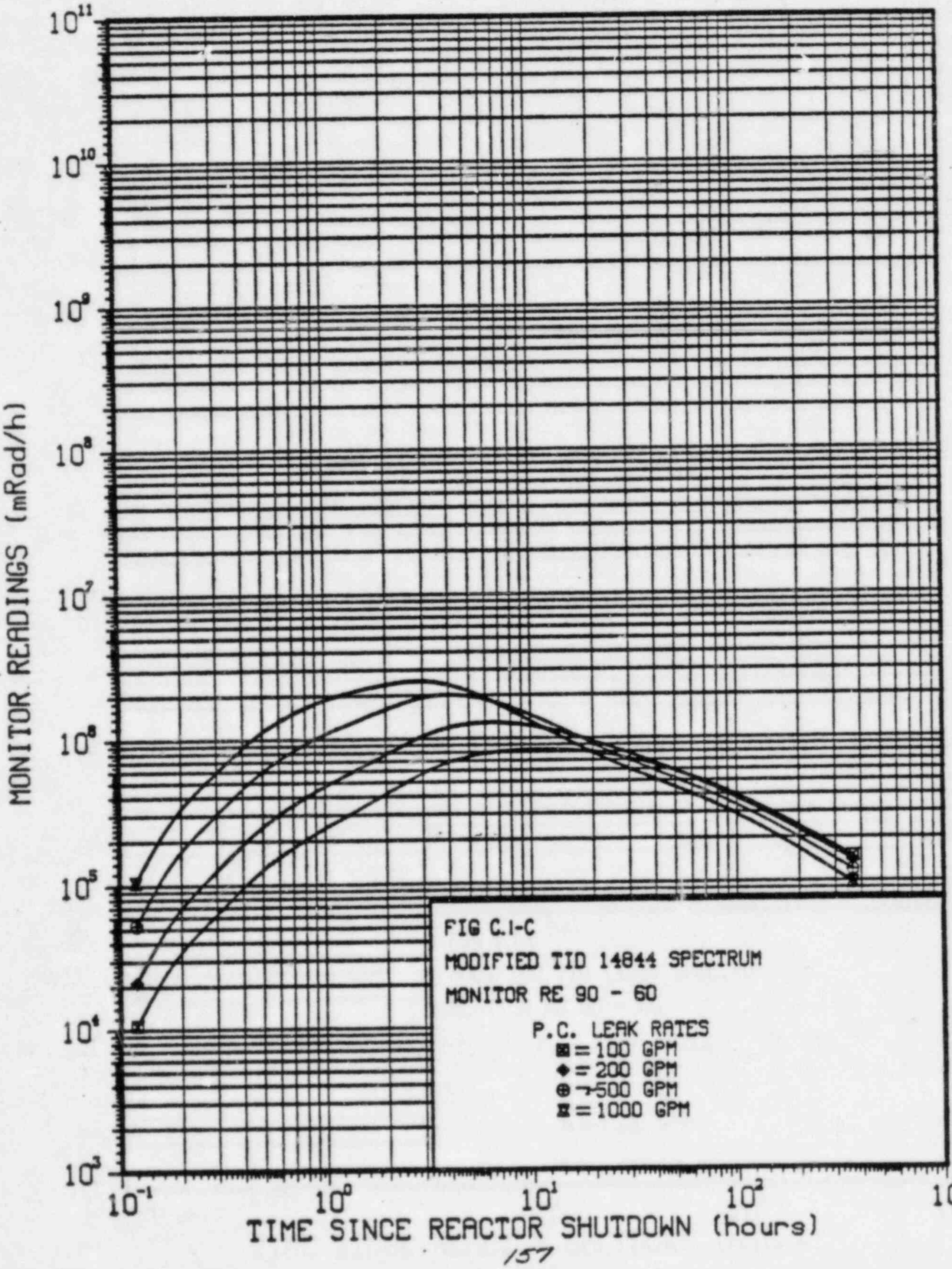
CONTAINMENT MONITOR READINGS vs. TIME

FOR DIFFERENT PRIMARY COOLANT LEAK RATES



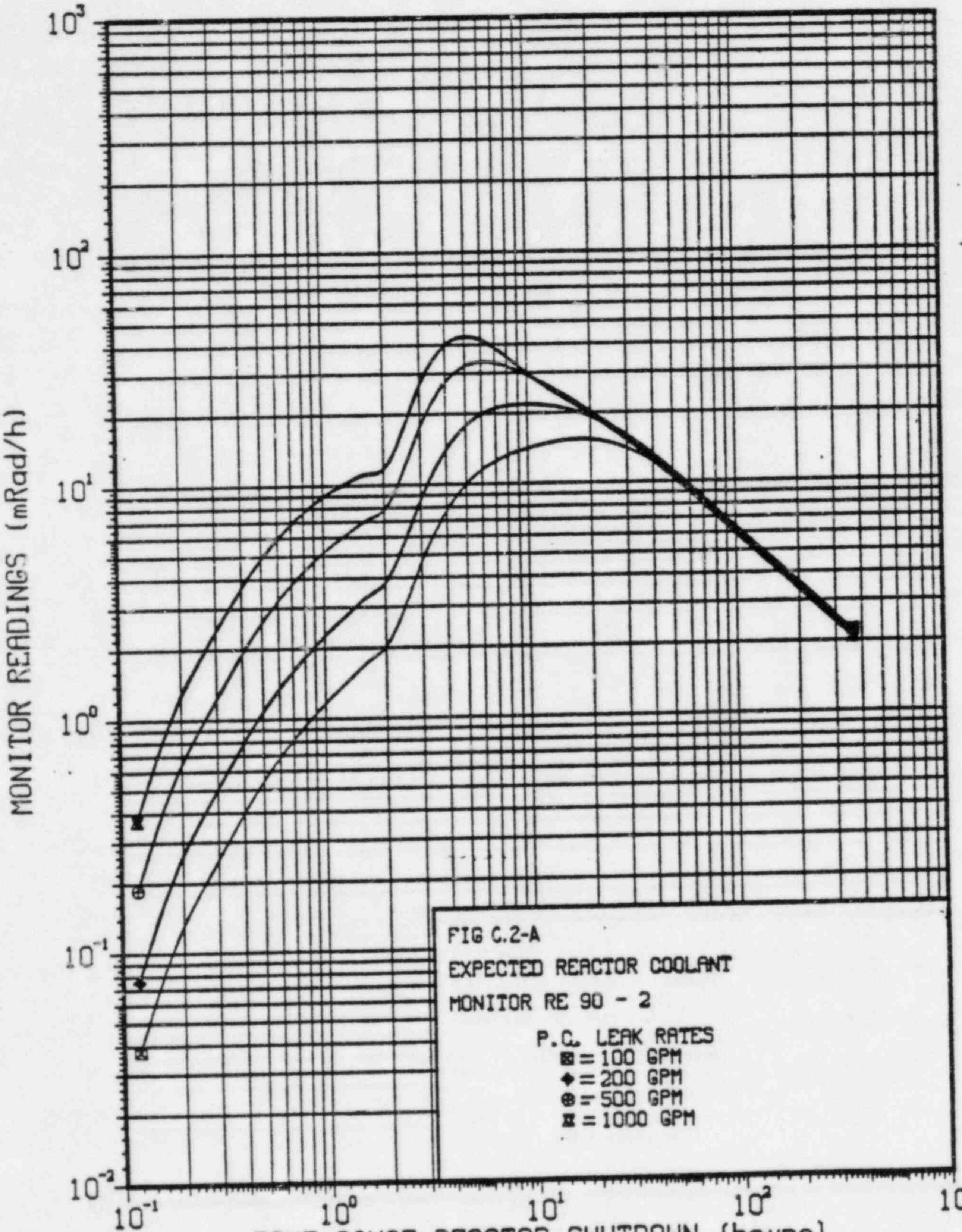
CONTAINMENT MONITOR READINGS vs. TIME

FOR DIFFERENT PRIMARY COOLANT LEAK RATES



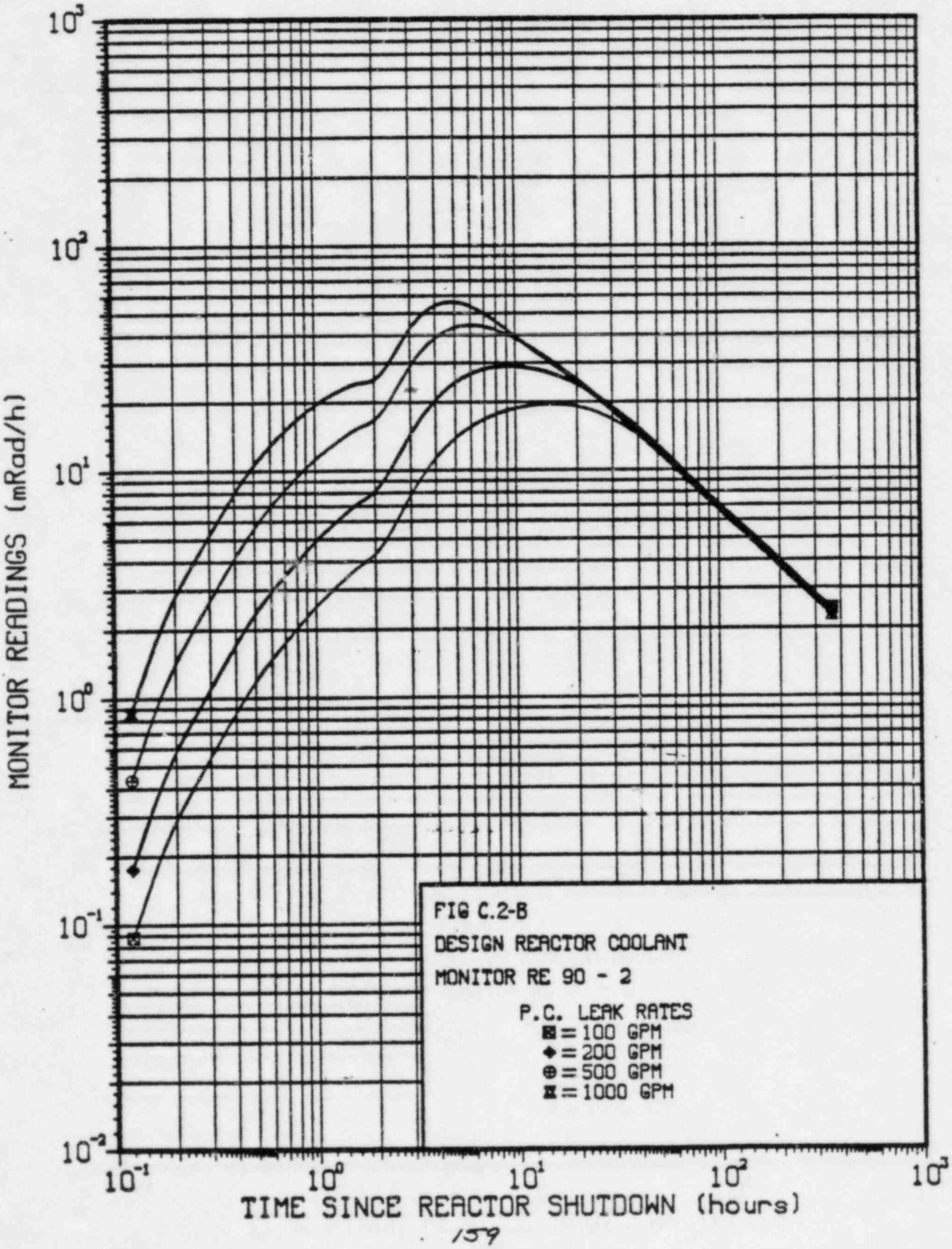
CONTAINMENT MONITOR READINGS vs. TIME

FOR DIFFERENT PRIMARY COOLANT LEAK RATES



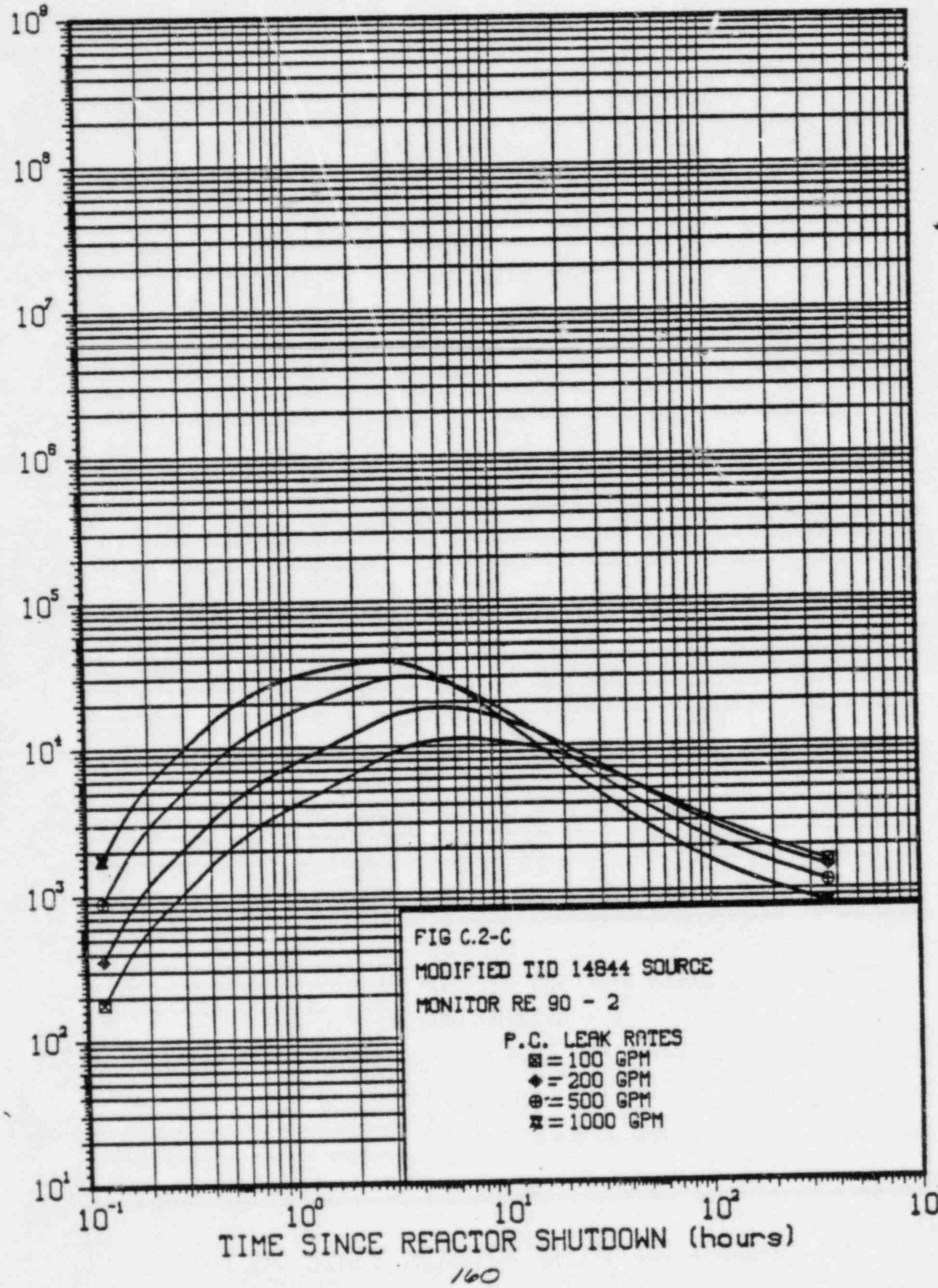
CONTAINMENT MONITOR READINGS vs. TIME

FOR DIFFERENT PRIMARY COOLANT LEAK RATES

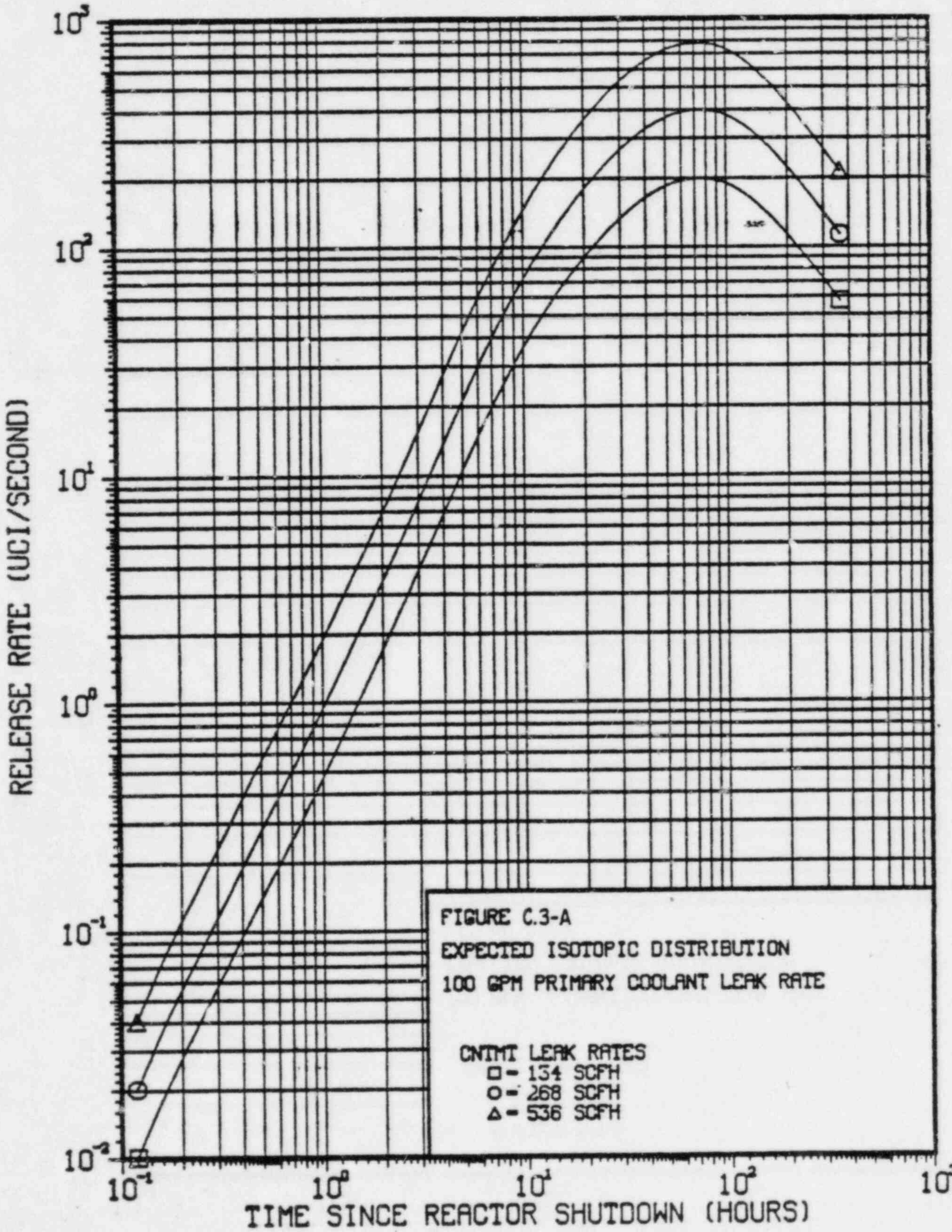


CONTAINMENT MONITOR READINGS vs. TIME

FOR DIFFERENT PRIMARY COOLANT LEAK RATES



NOBLE GAS RELEASE RATE
FOR DIFFERENT CONTAINMENT LEAK RATES



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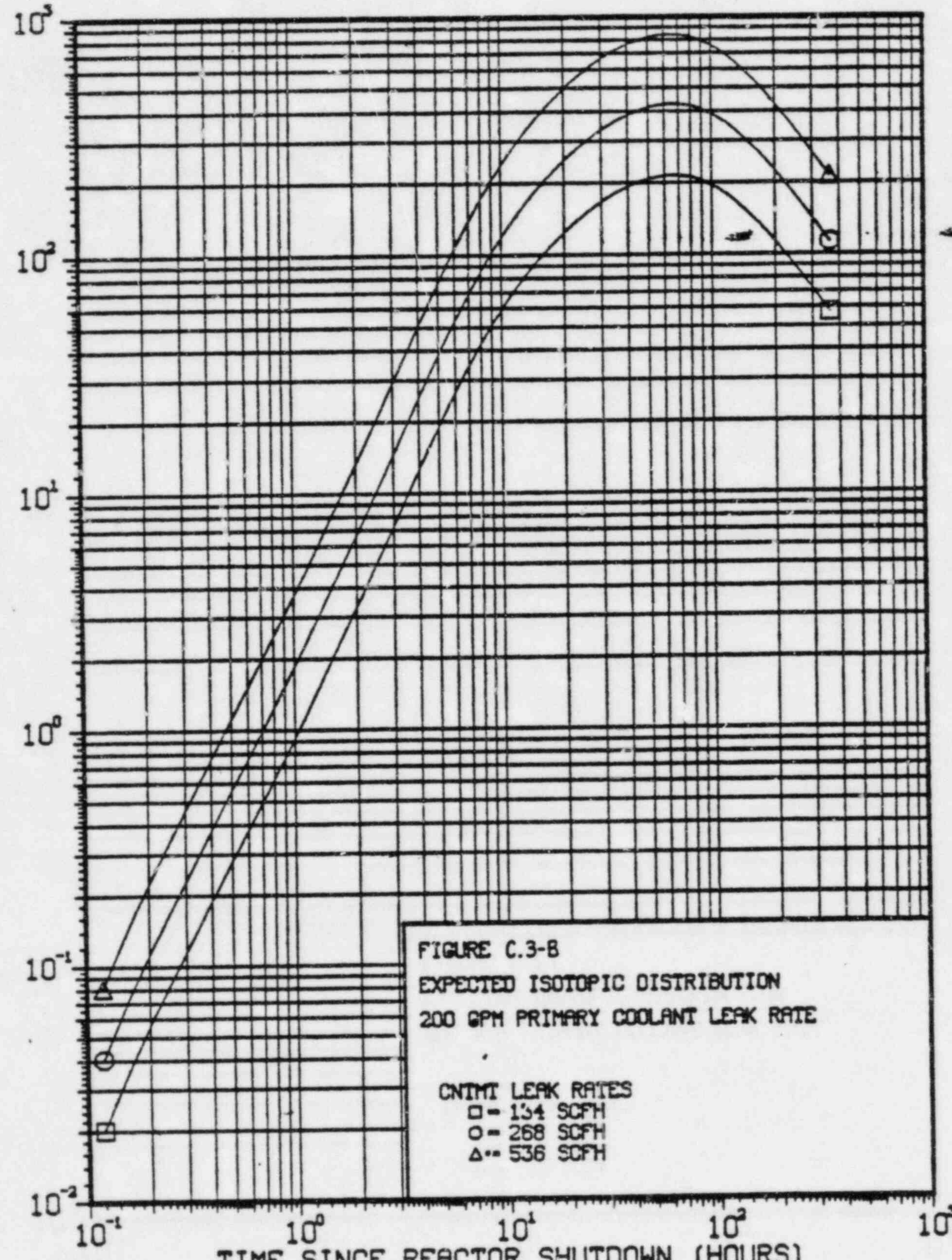

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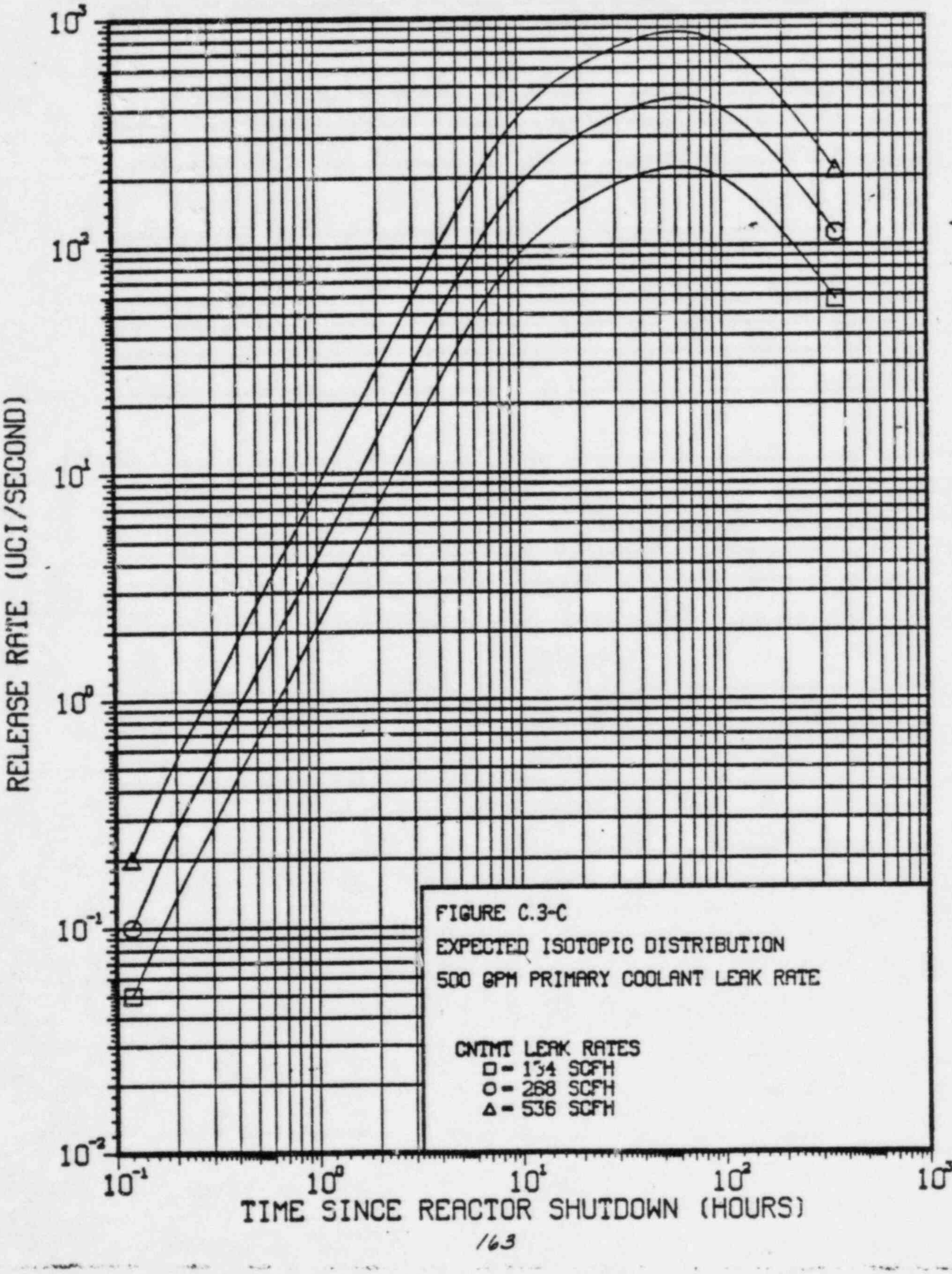
FOR DIFFERENT CONTAINMENT LEAK RATES



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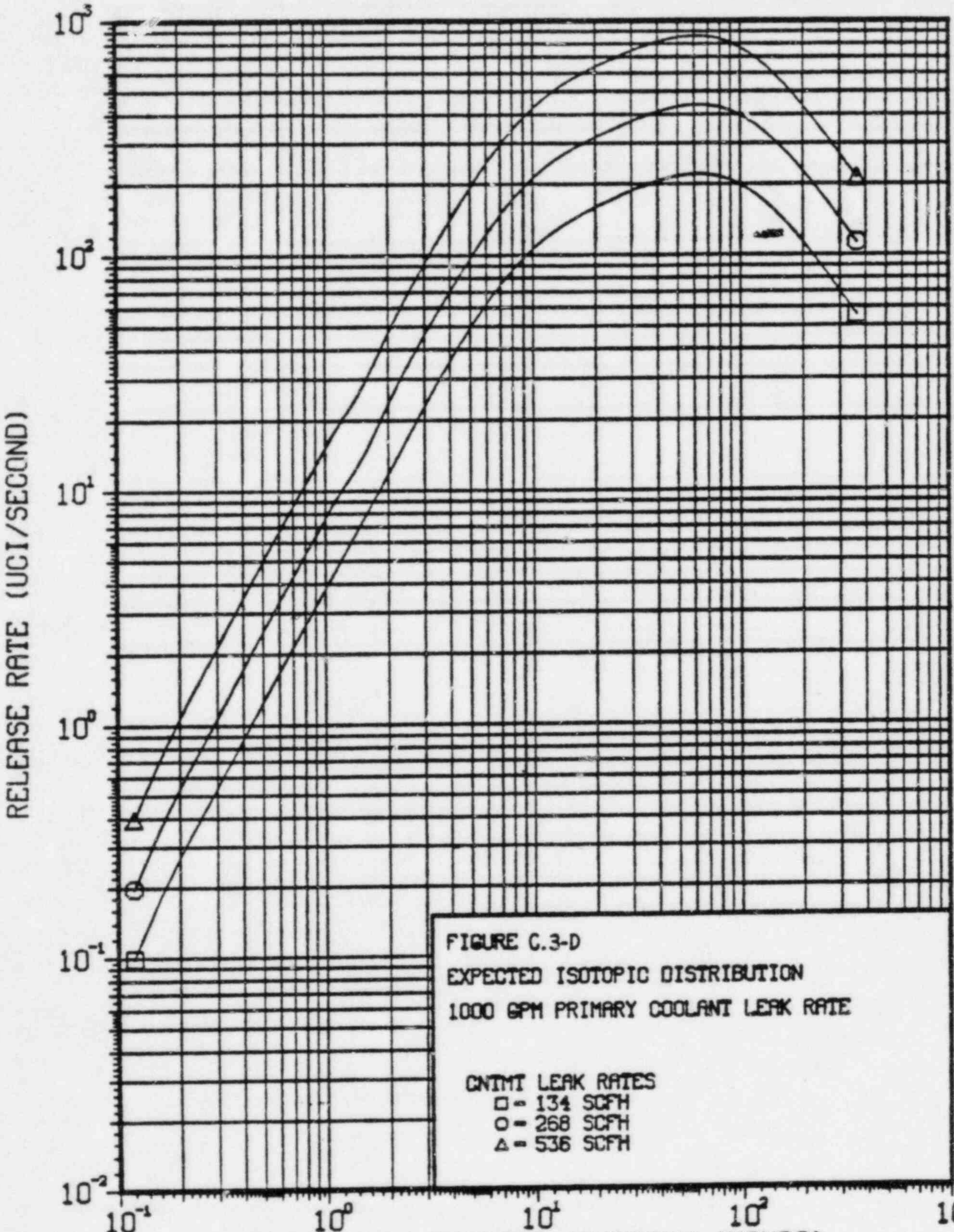

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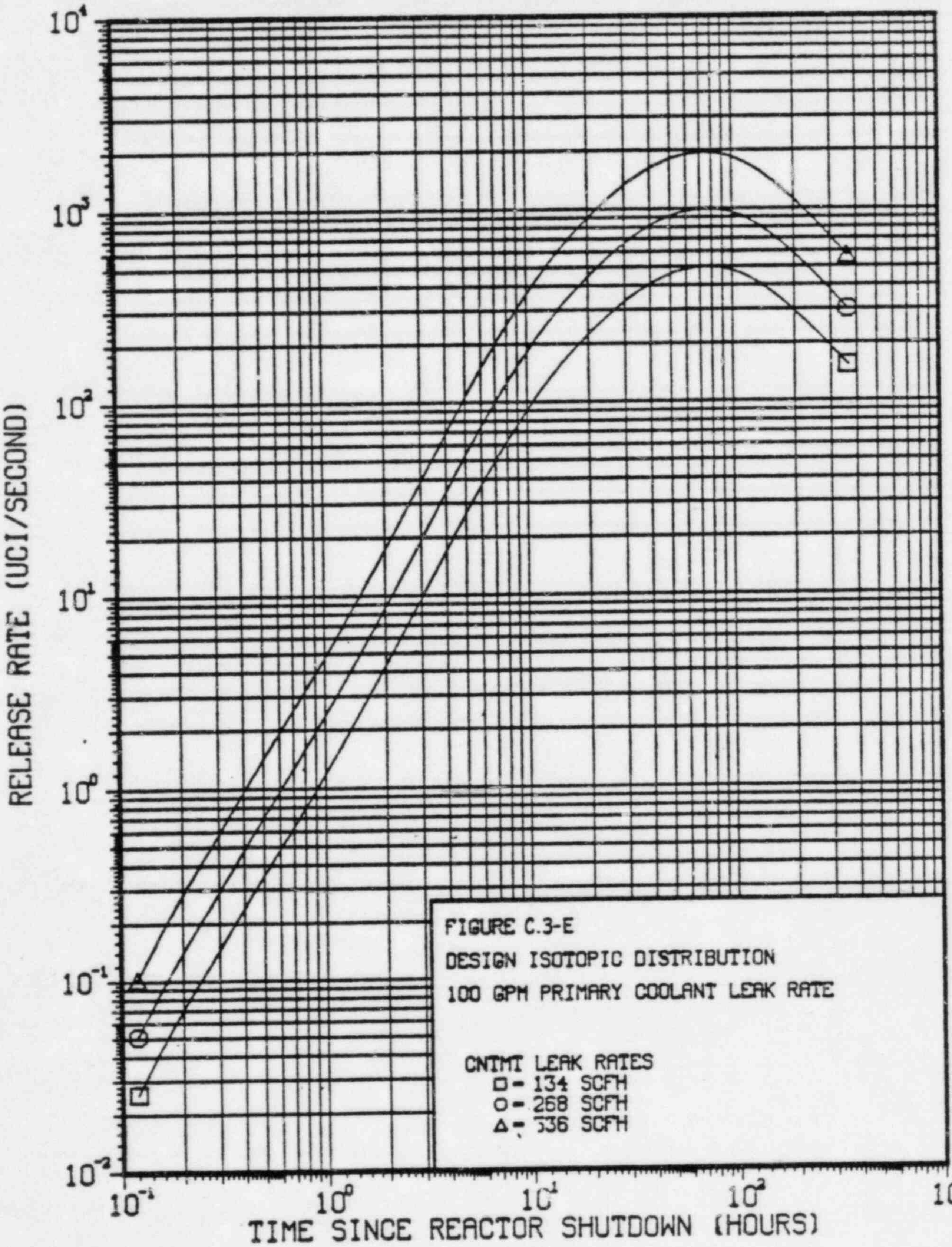

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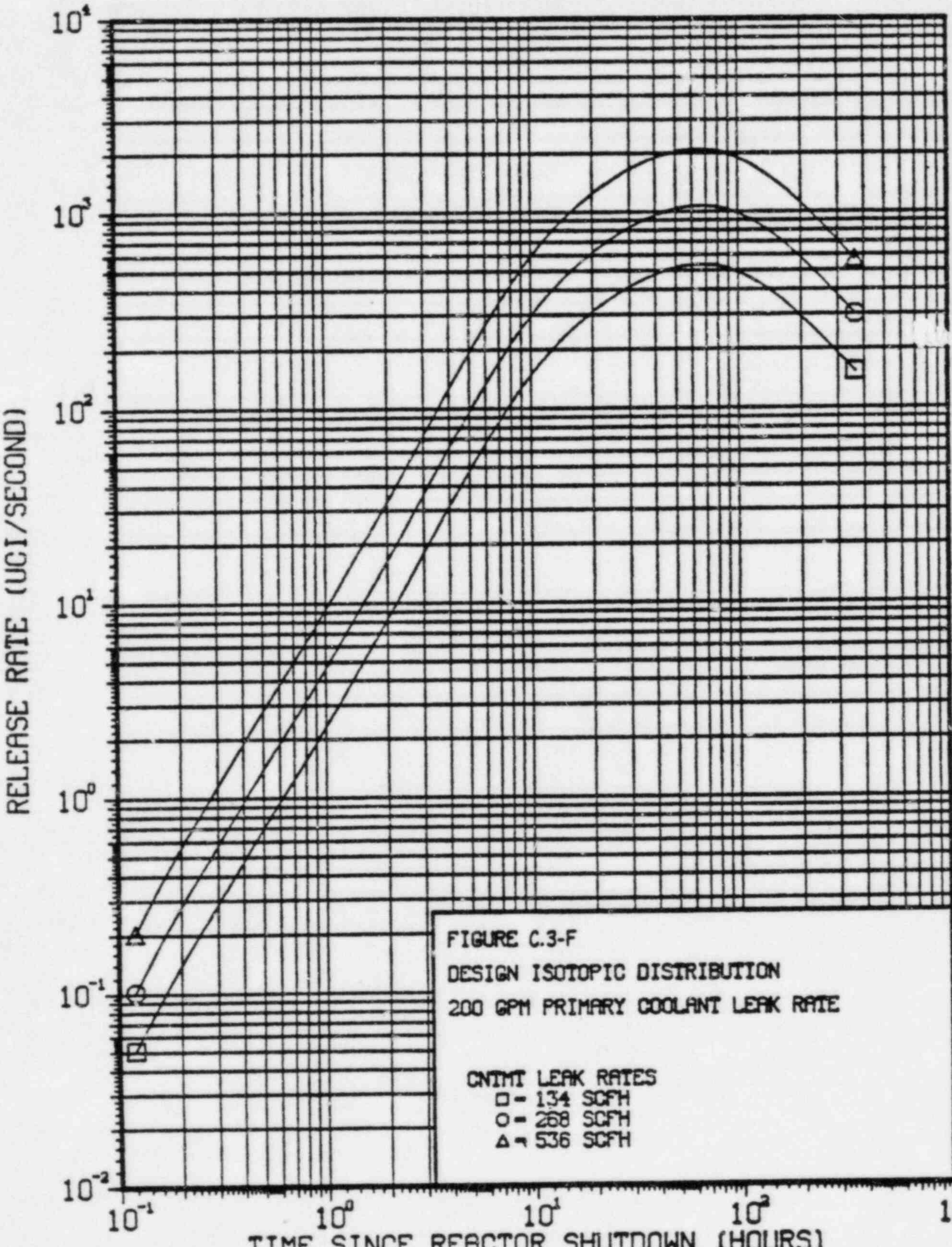

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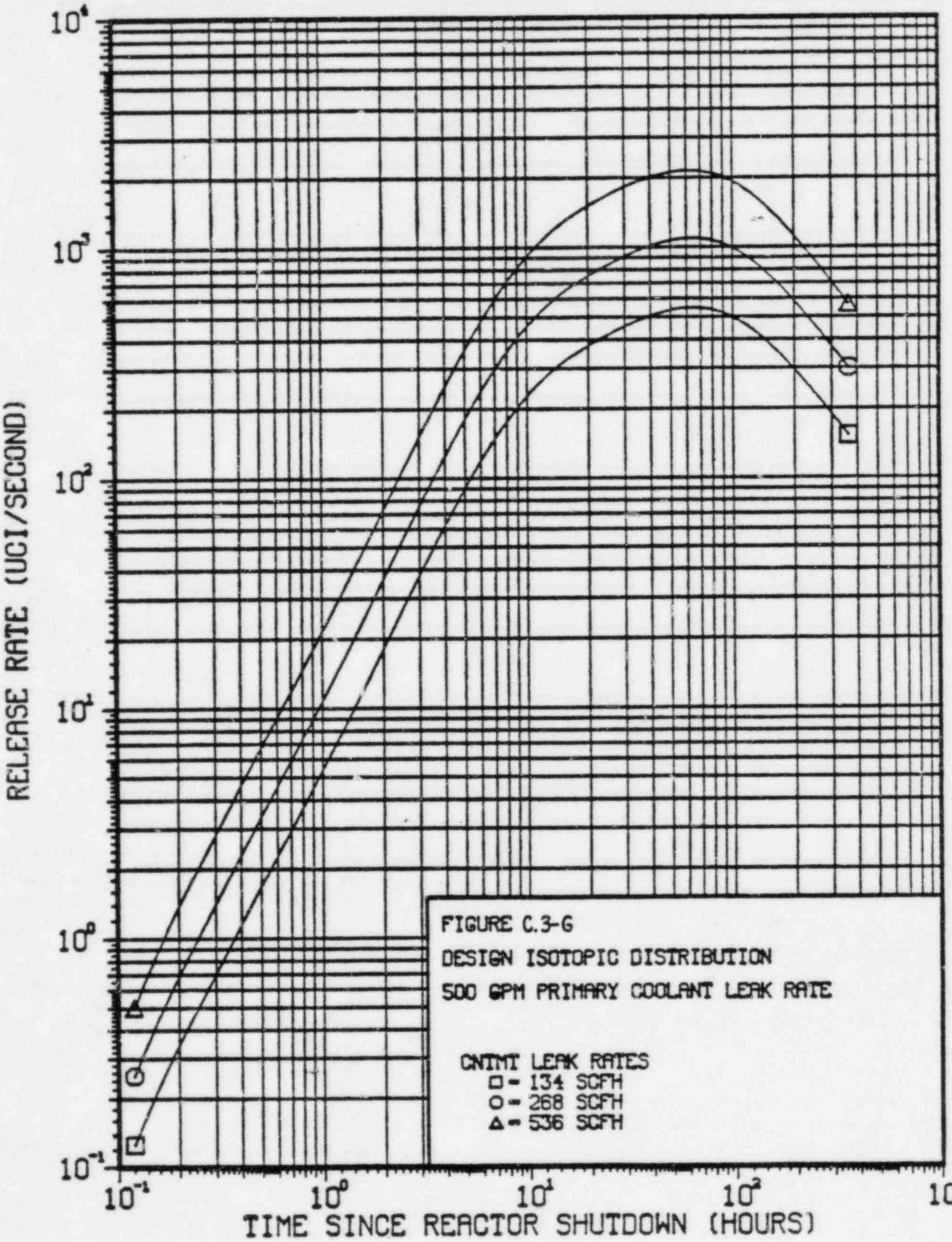

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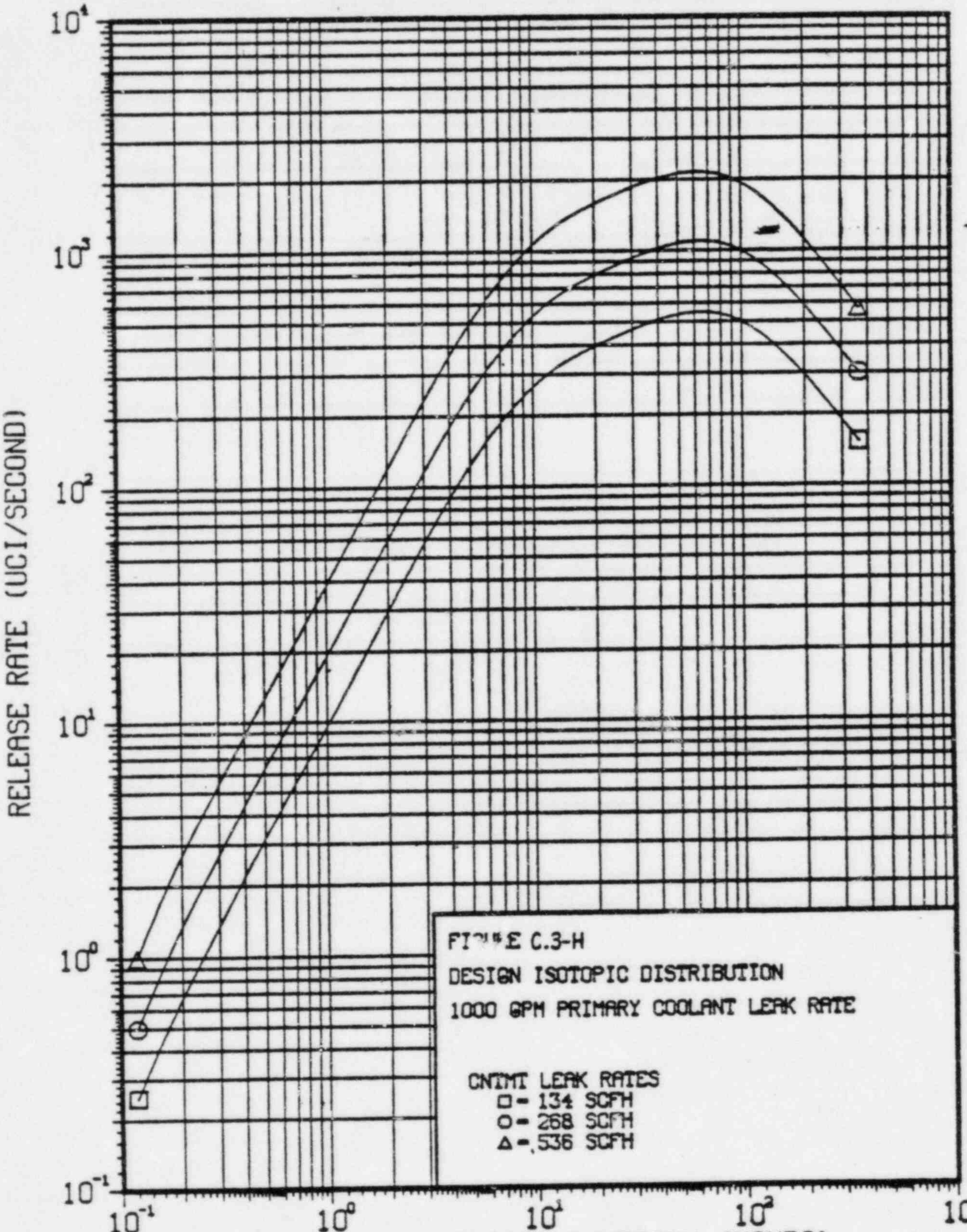

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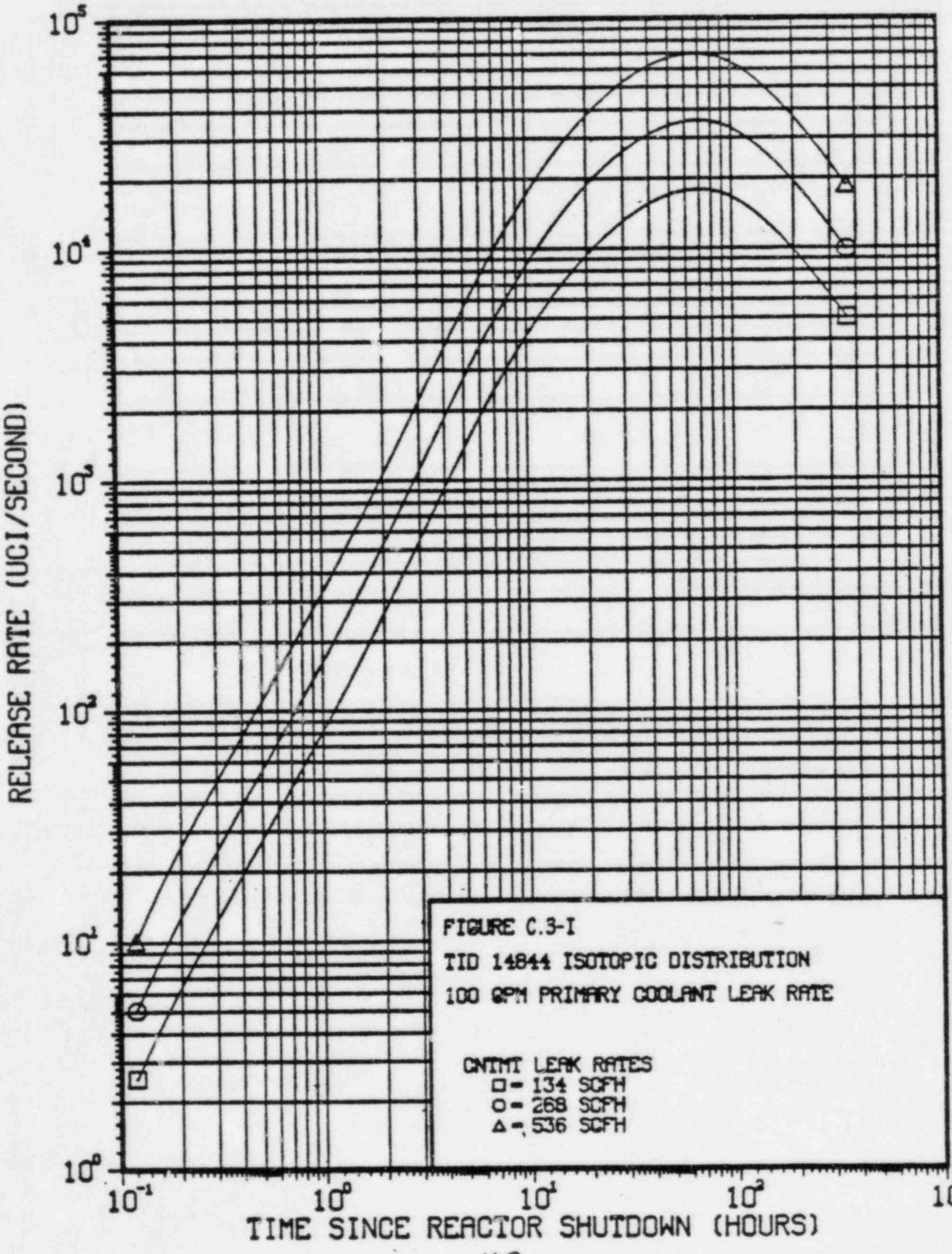
FOR DIFFERENT CONTAINMENT LEAK RATES



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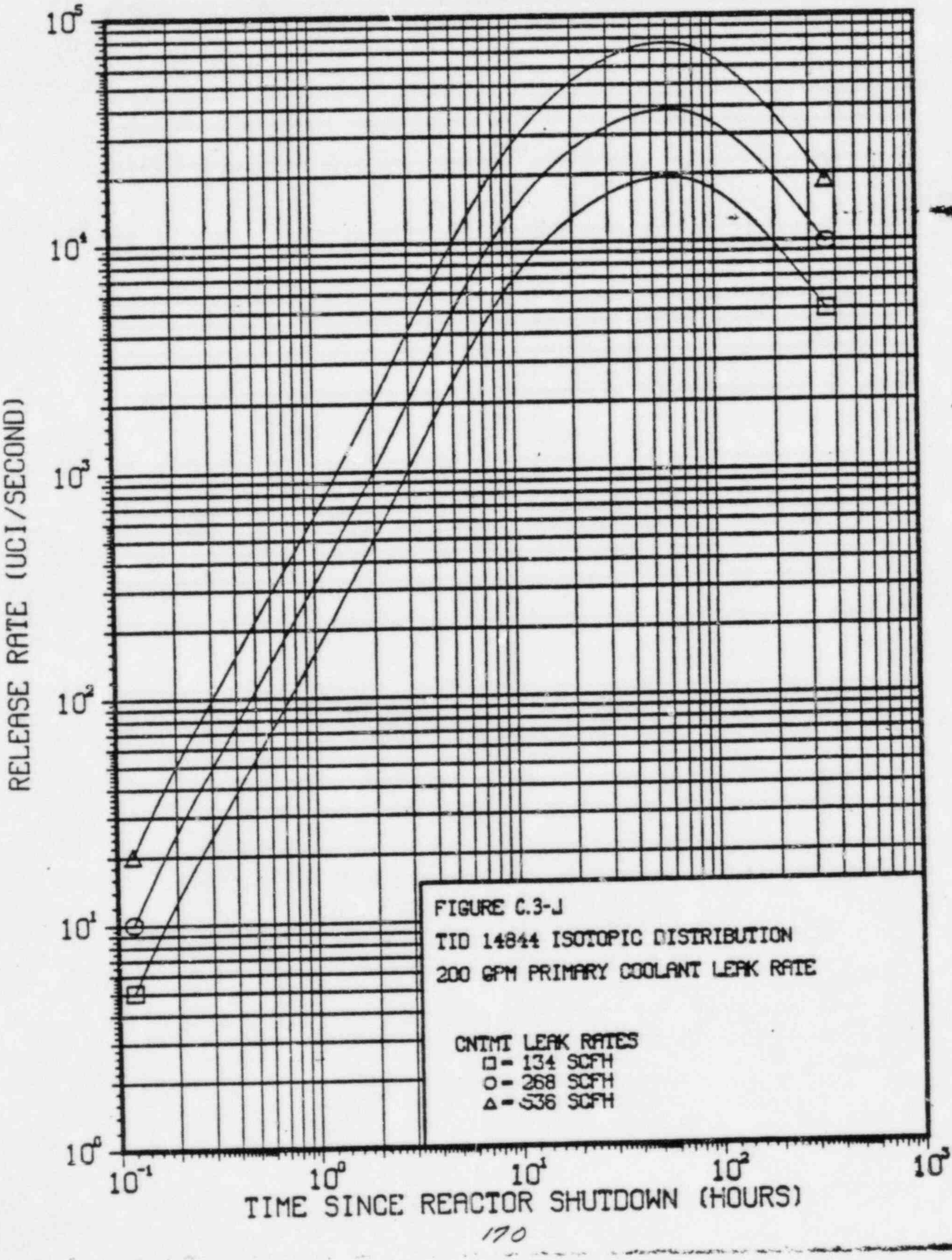
FOR DIFFERENT CONTAINMENT LEAK RATES



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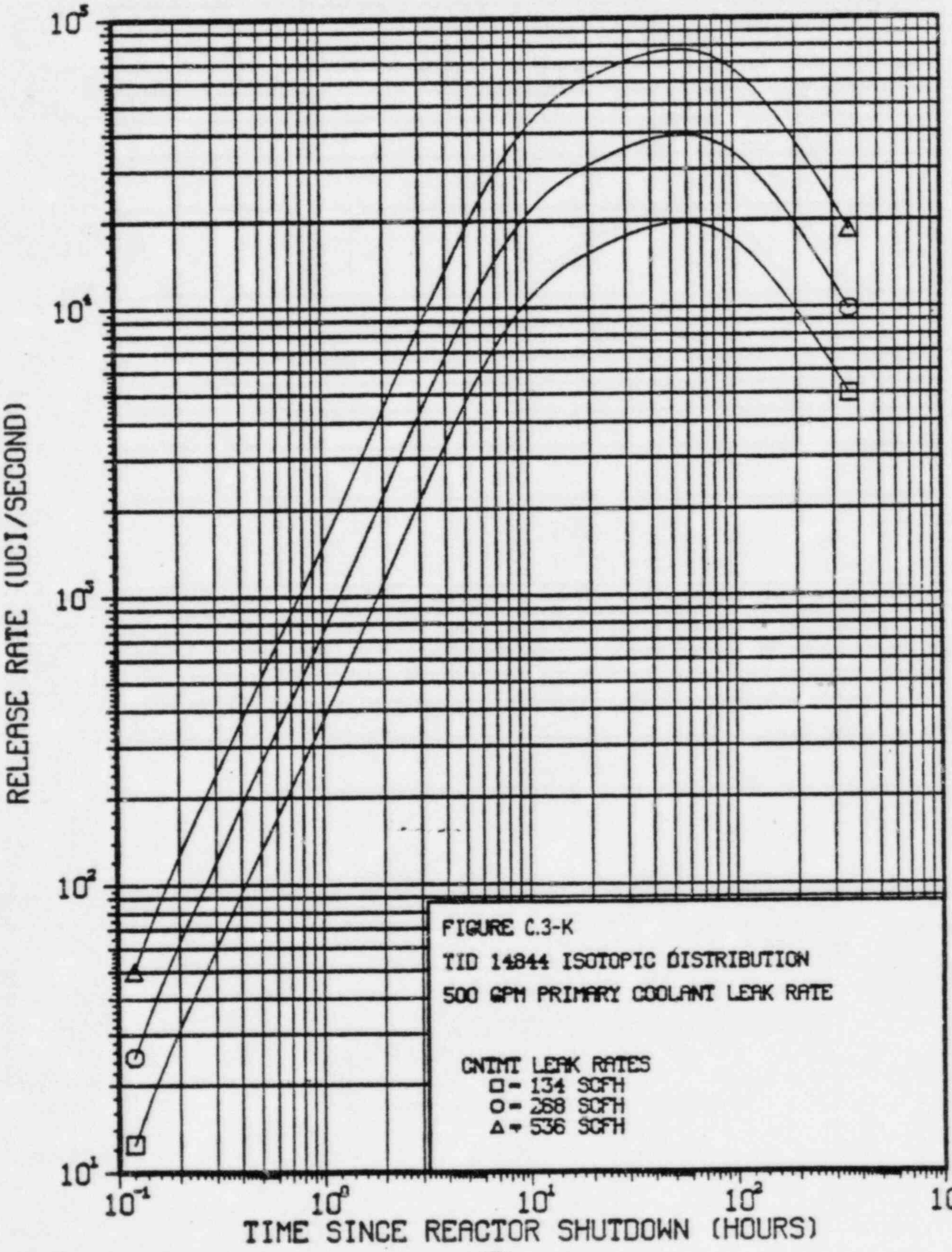
FOR DIFFERENT CONTAINMENT LEAK RATES



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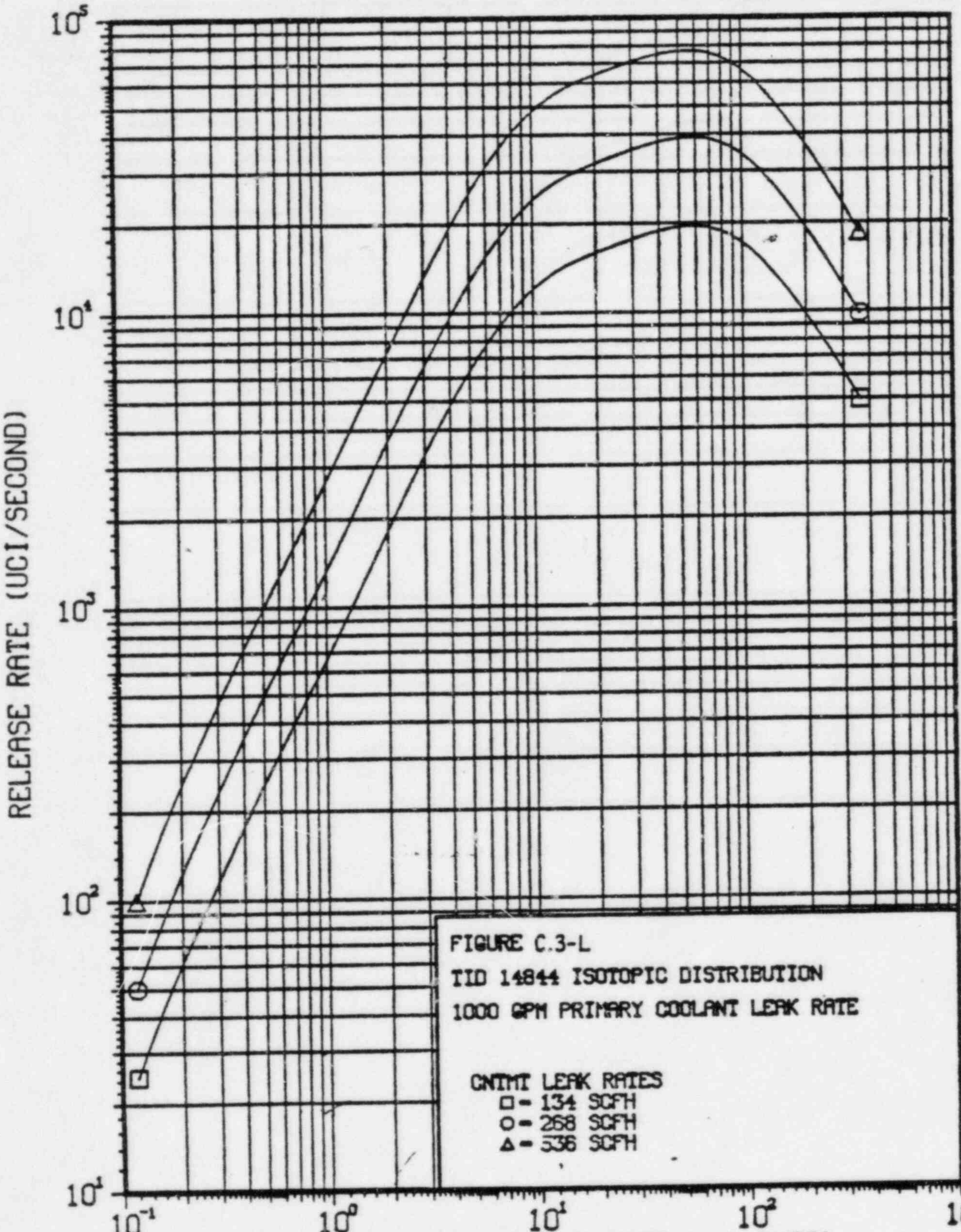
FOR DIFFERENT CONTAINMENT LEAK RATES



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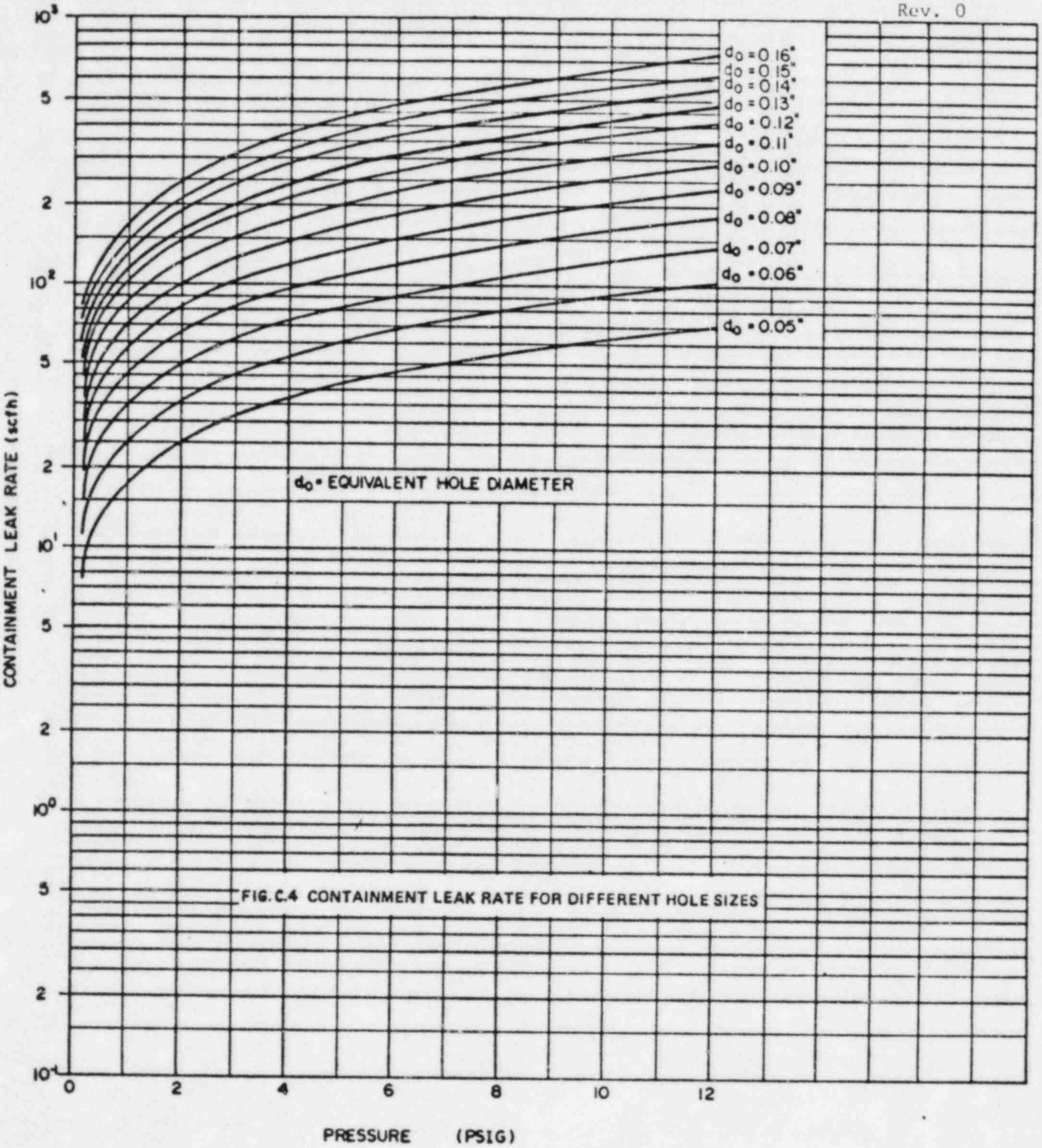
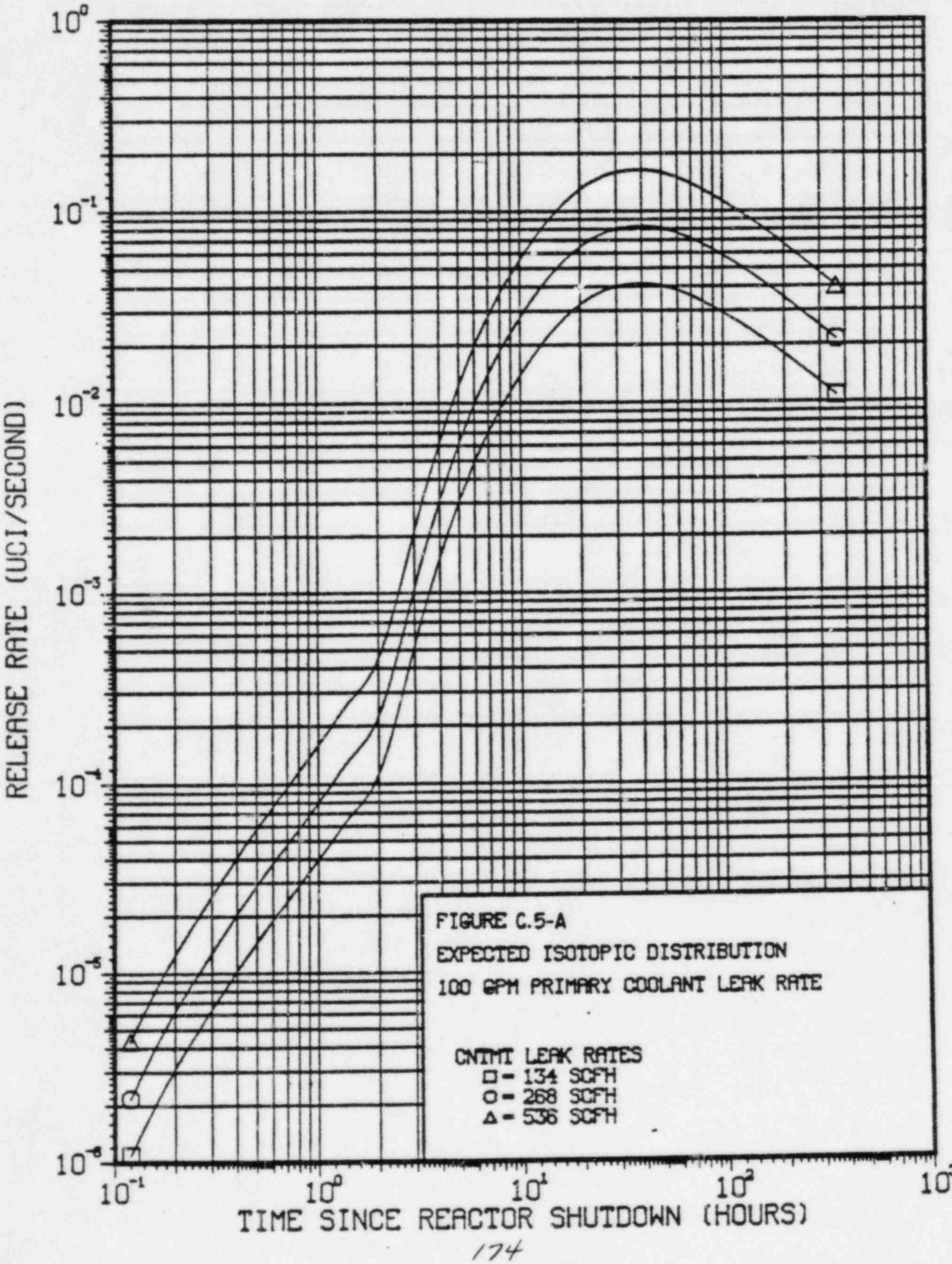
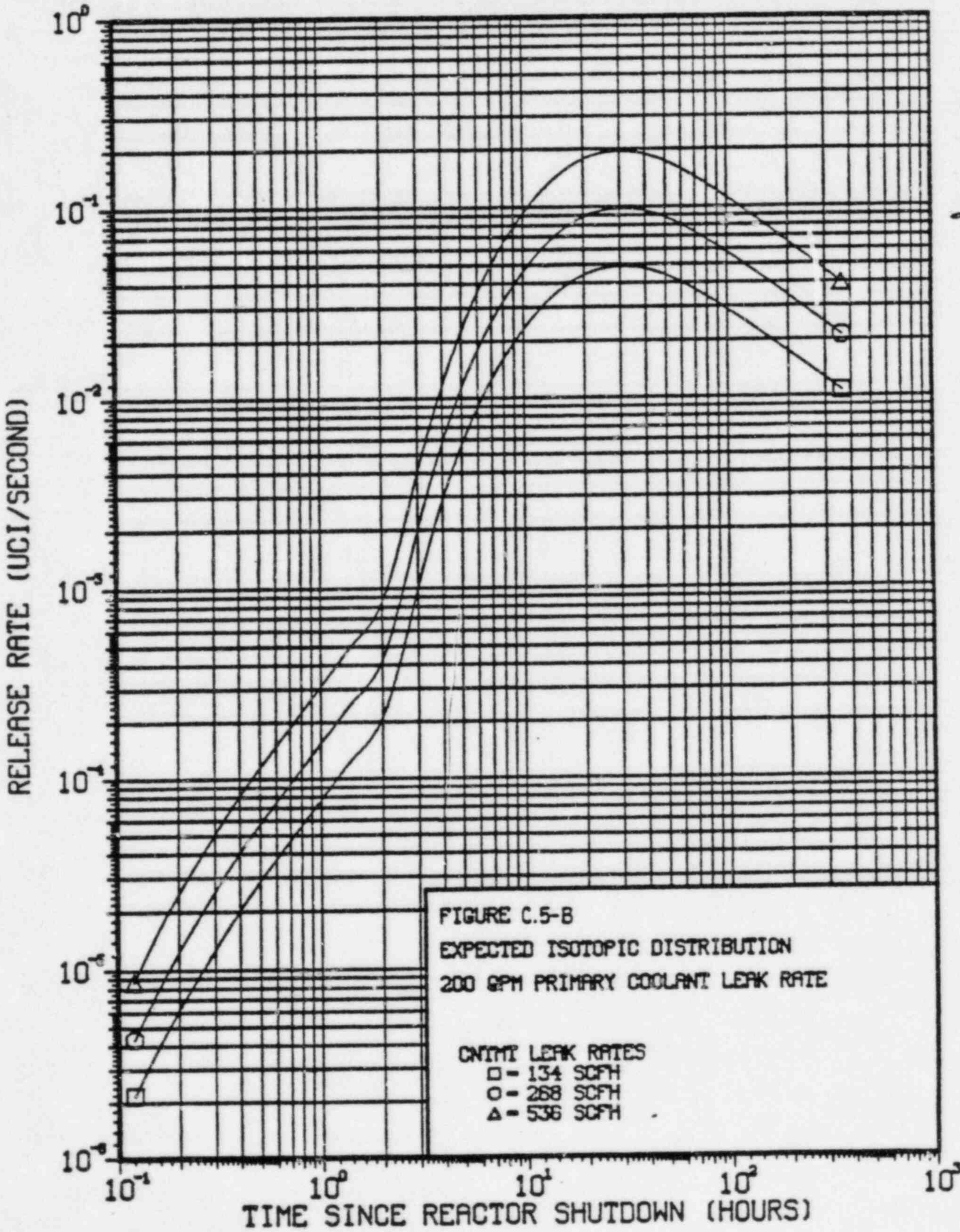


FIG. C.4 CONTAINMENT LEAK RATE FOR DIFFERENT HOLE SIZES

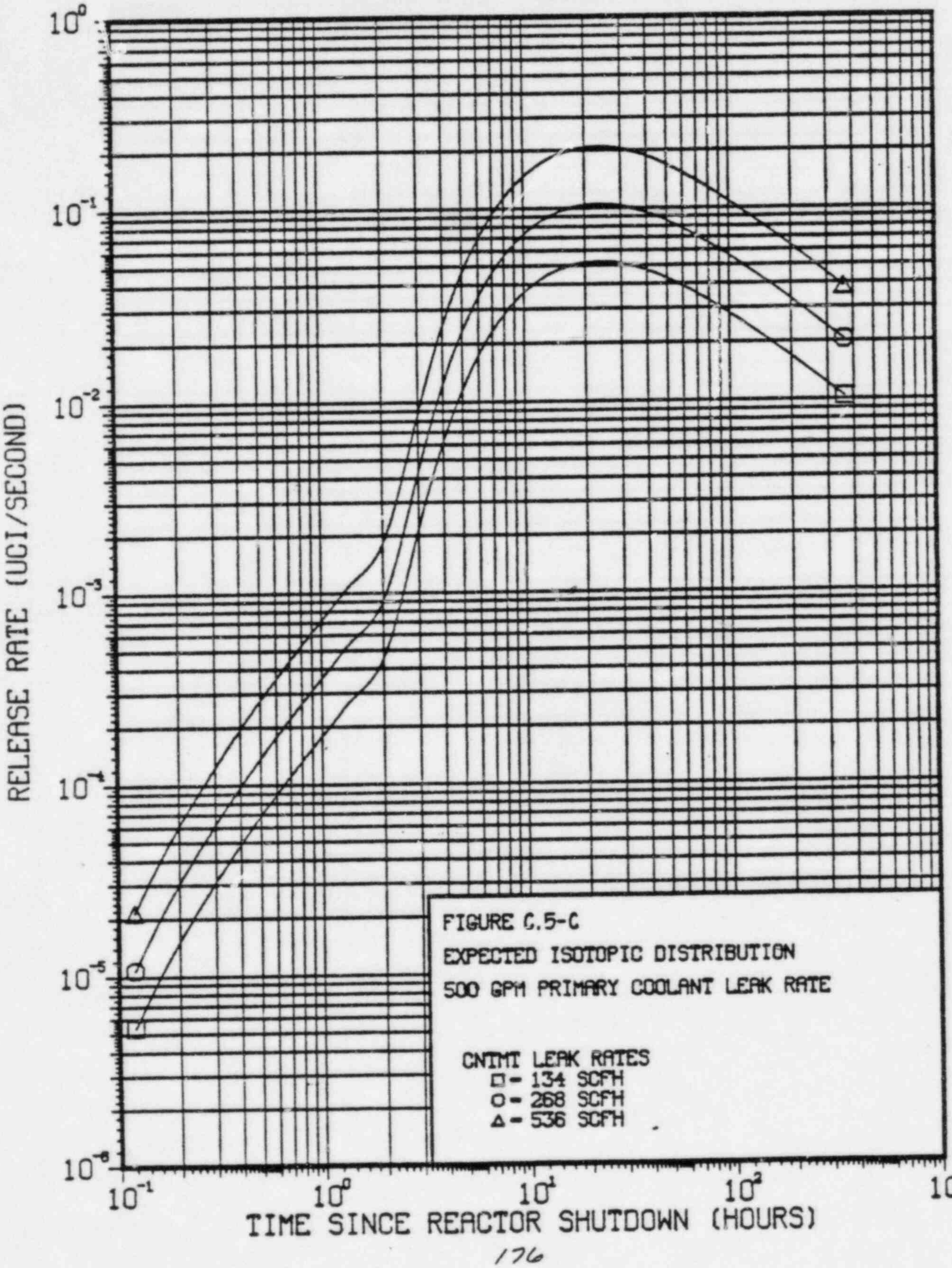
IODINE RELEASE RATE
FOR DIFFERENT CONTAINMENT LEAK RATES



IODINE RELEASE RATE
FOR DIFFERENT CONTAINMENT LEAK RATES



IODINE RELEASE RATE
FOR DIFFERENT CONTAINMENT LEAK RATES



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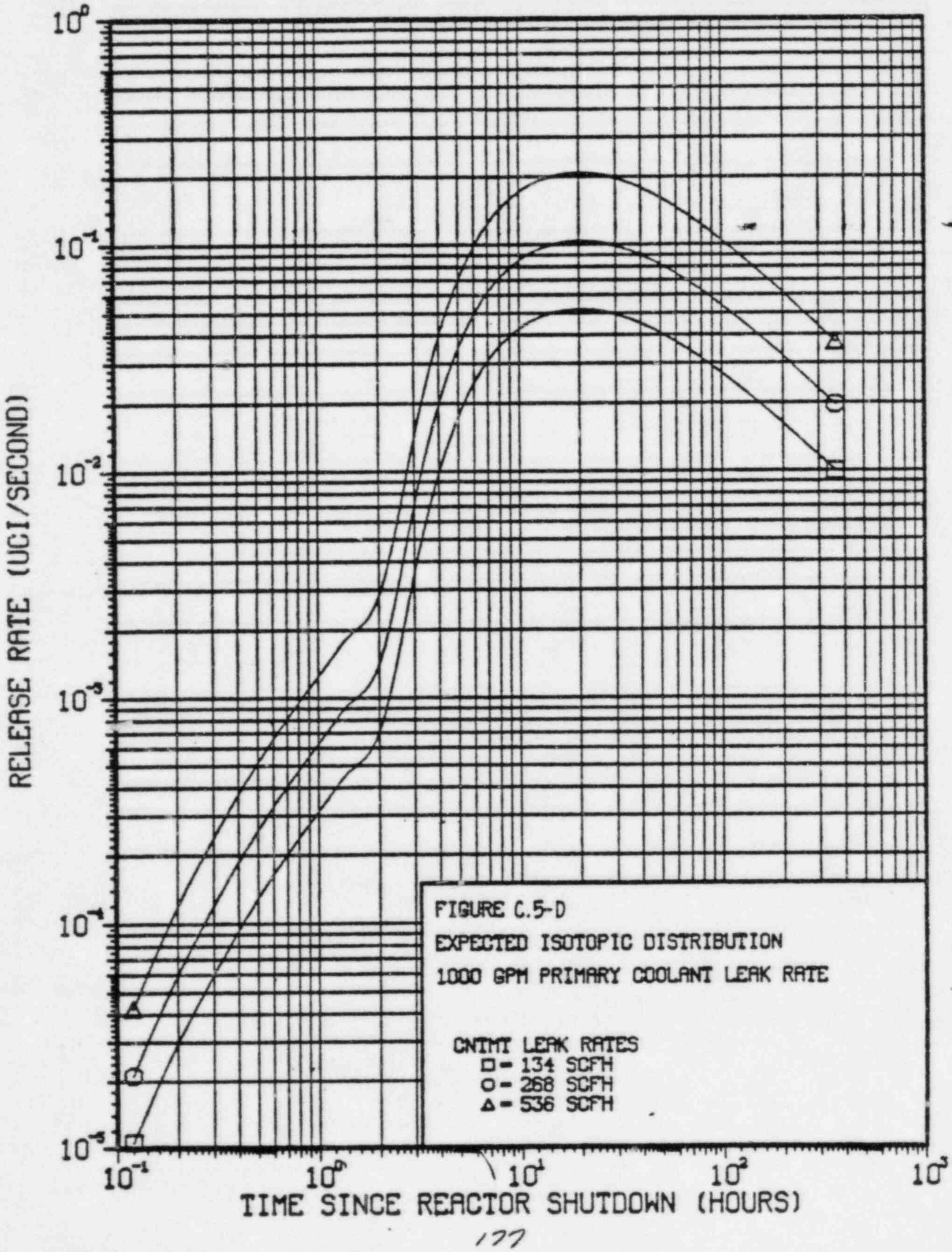

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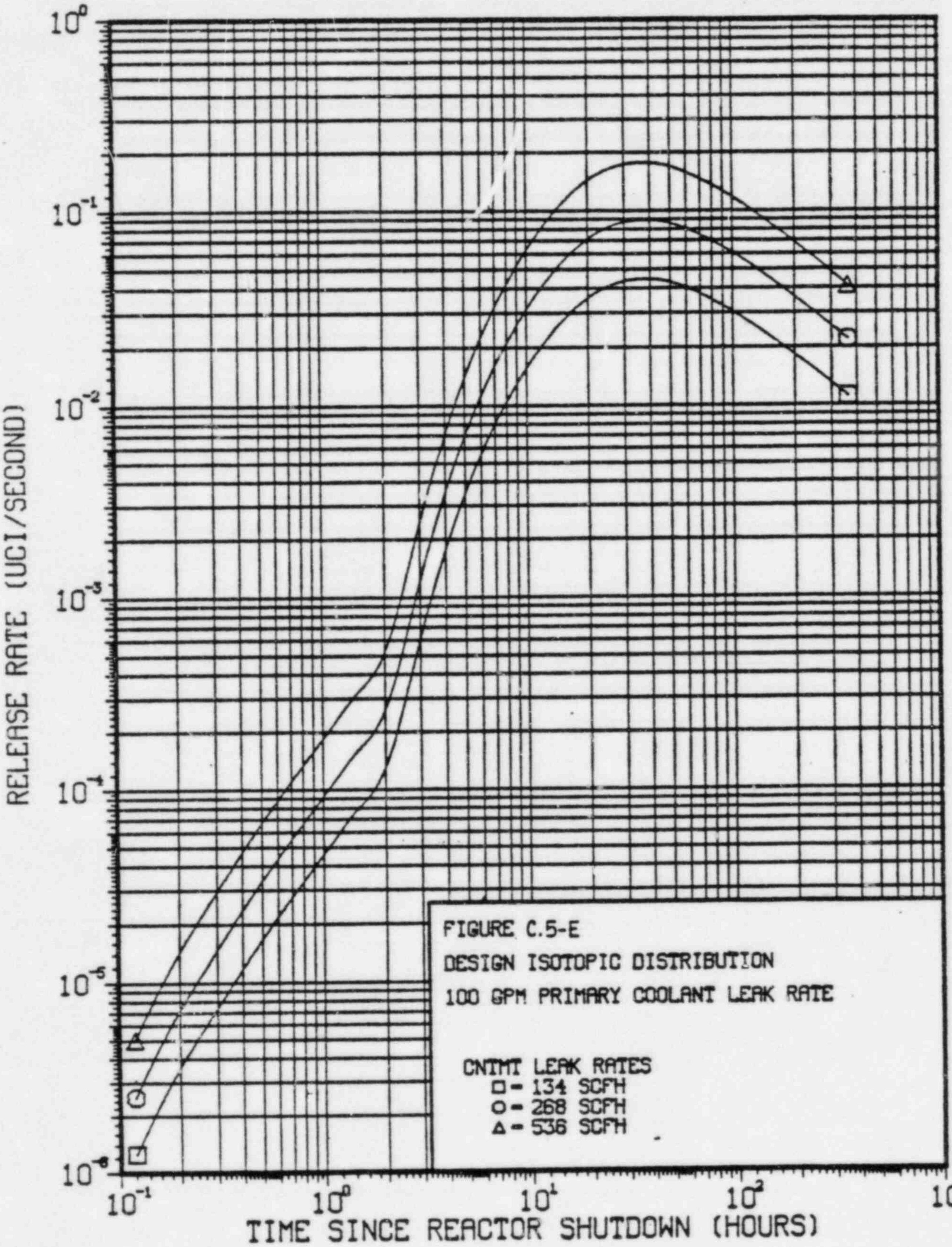
FOR DIFFERENT CONTAINMENT LEAK RATES



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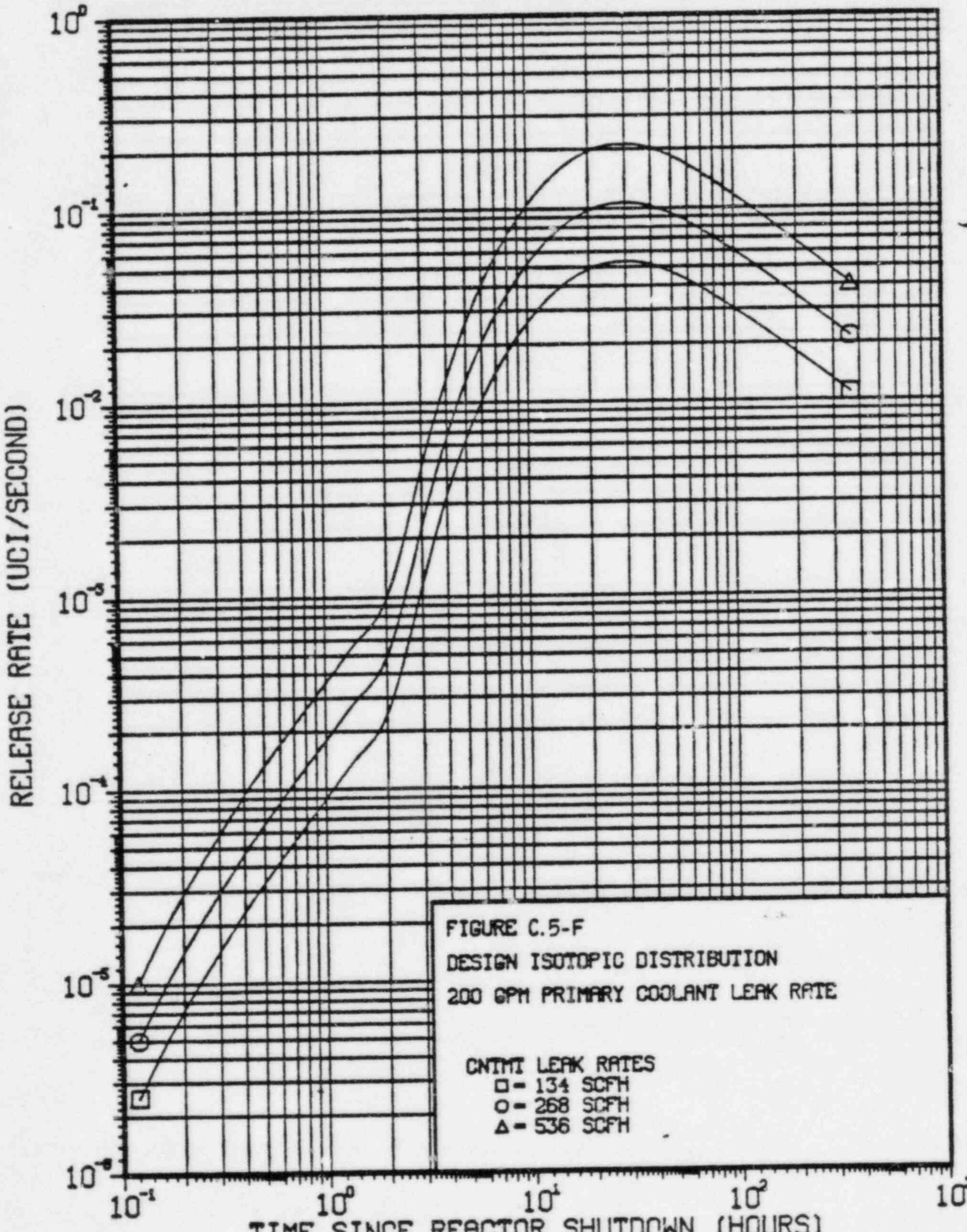

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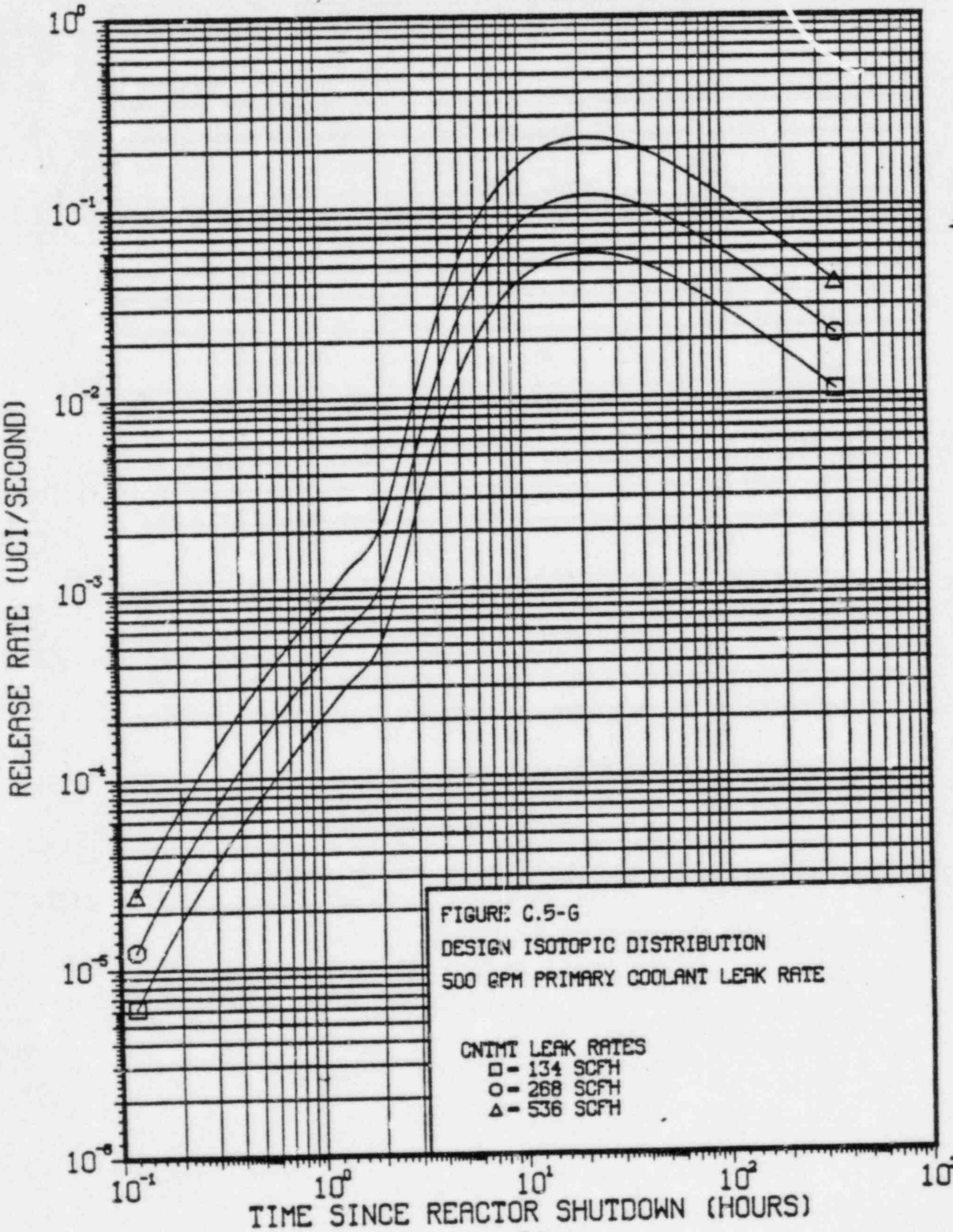

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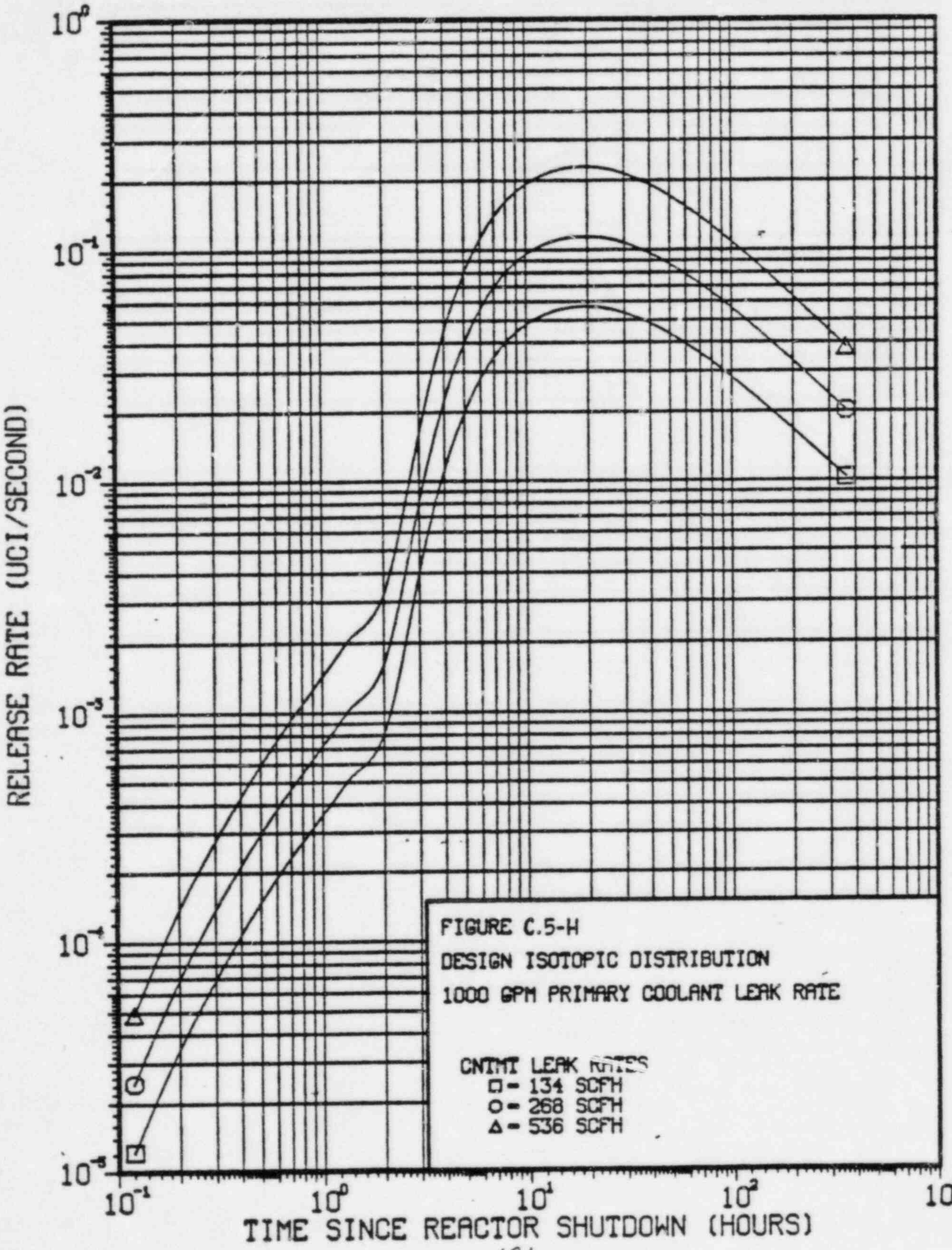

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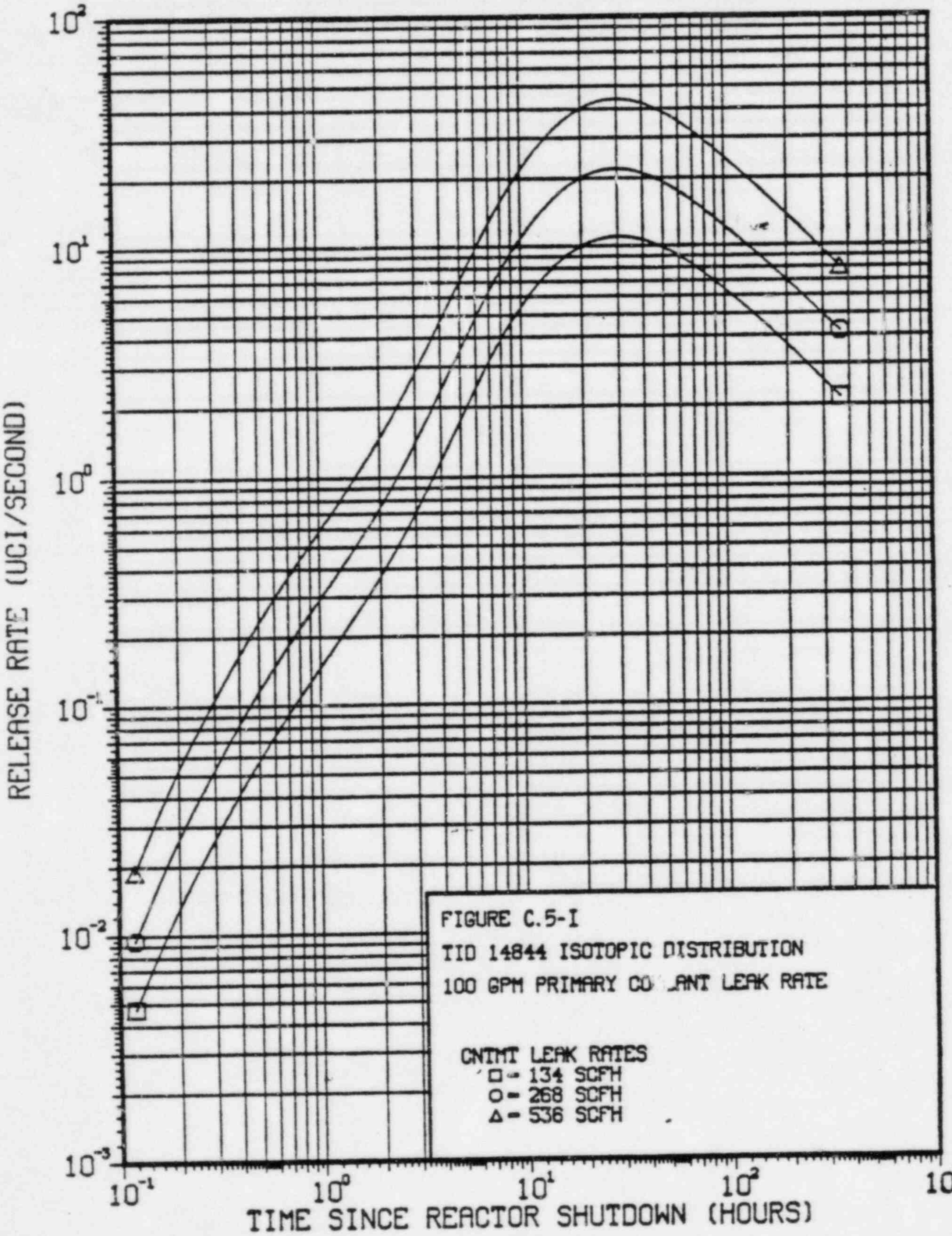

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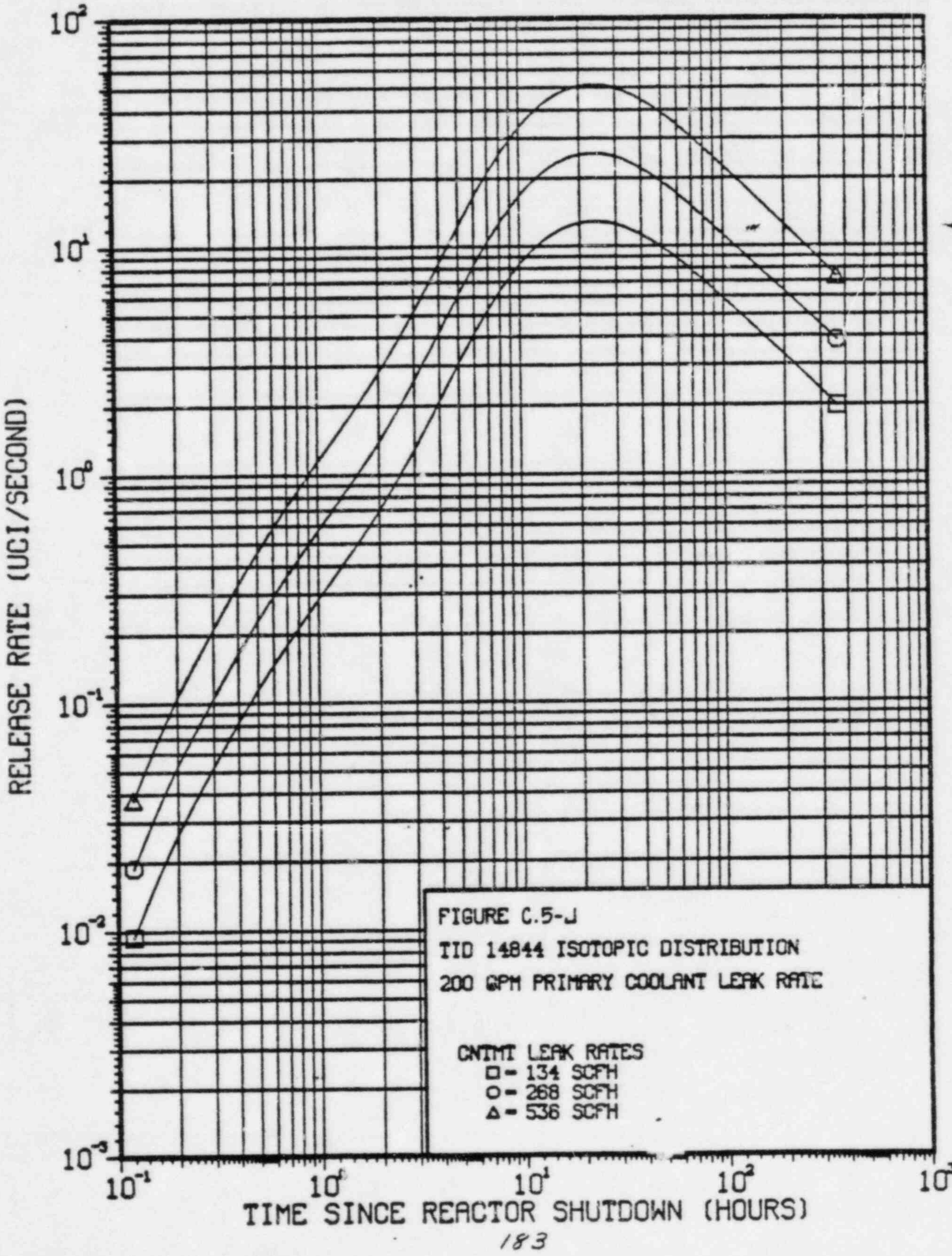

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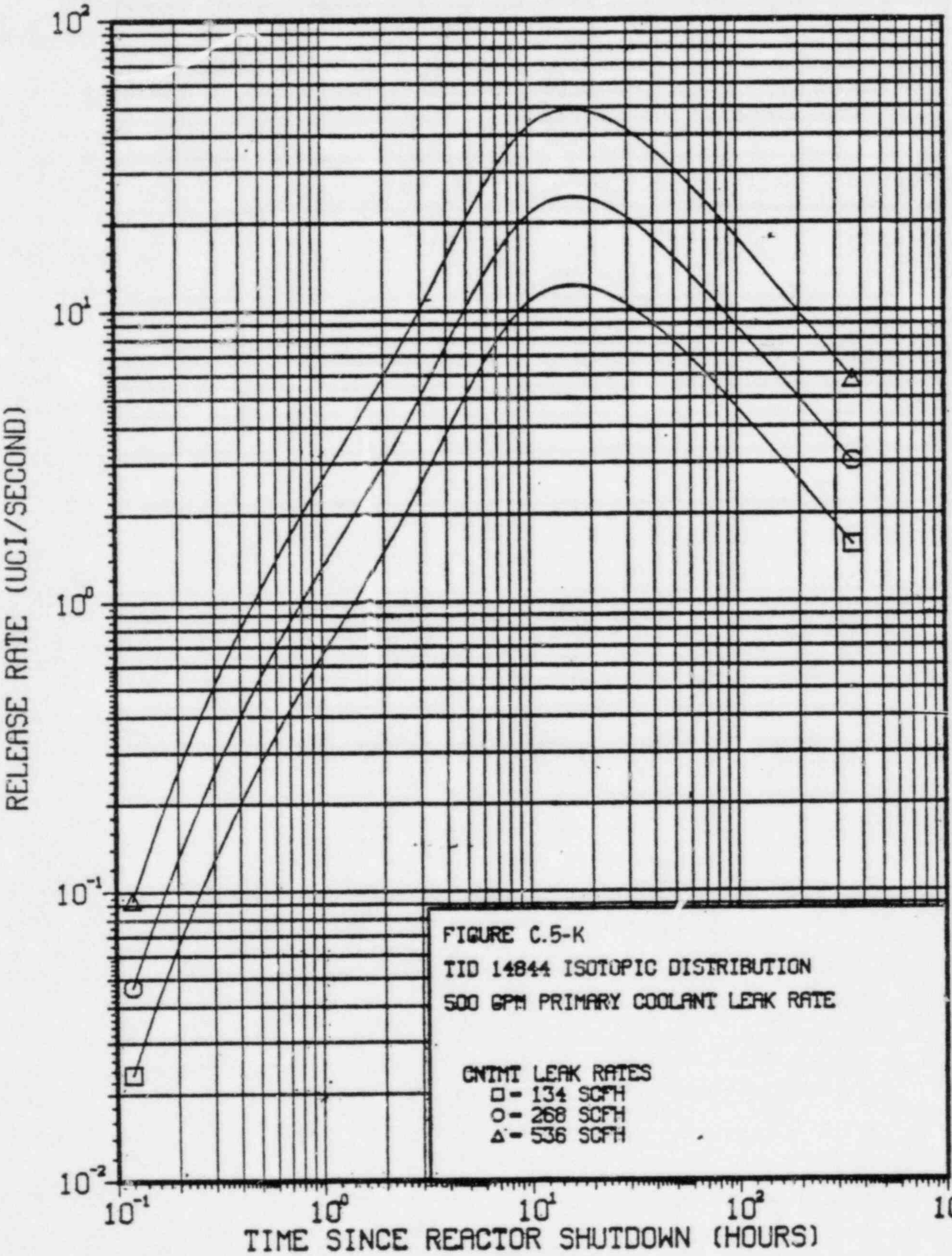

IODINE RELEASE RATE
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