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ANALYSIS OF LEAK PROPAGATION FOR DBL SELECTIONS OF MUNJU STEAM GENERATORS

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Abstract

Analyses of leak propagation have been performed in order to support the selection of a design basis leak (DBL) for sodium-water reaction accidents in steam generators of the Japanese prototype fast breeder reactor, MONJU.

For the purpose of estimating the possible maximum leak rate due to the failure propagation, a computer code, LEAP (Leak Enlargement And Propagation), has been developed. This code analyzes the failure propagation process taking account of data from sodium-water reaction experiments throughout the world.

This paper describes the model and structure of the LEAP code, a validation study, and results of the LEAP calculation for the MONJU steam generators and it has been concluded that the sodium-water reaction accidents can be terminated before the leak rate exceeds the DBL (one plus three DEG's) with the present detection system and safety devices.

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1. INTRODUCTION

In an LMFBR steam generator, the chemically reactive sodium and water are used as the heat transfer media. For this reason, the safety of the steam generator, particularly the protection for sodium-water reactions, is a very important problem. It is necessary to confirm the reliability of an LMFBR plant with regard to its availability. A research program of steam generator safety and reliability has been conducted for more than ten years in Japan, and results are reflected in the steam generator design.

Generally, the growth behavior of the sodium-water reaction accident can be described as follows: A leak initiates as a micro-leak due to a faulty weld or another imperfection, and self-enlargement would increase the leak rate to a small or intermediate leak level. Moreover, if no action were taken against the leak, it might grow to a large leak level due to failure propagation effects.

In the practical plant steam generator system, the leak could be detected in the early stage and the relative operations, including an emergency water blowdown, could prevent failure propagation. However, the time required for operations might allow failure propagation to a certain degree; thus the extent of failure propagation should be estimated adequately for the steam generator design. This design basis leak (DBL) is the basis of the steam generator system design and must be chosen prudently with regard to results of experimental studies of the sodium-water reactions and the ability of related equipment design.

Many examinations have been performed on the sodium-water reaction phenomena from the micro-leak to the large leak region using the sodium water reaction test facilities: the Large Leak Sodium-Water Reaction Test Rig (SWAT-1), the Small Leak Sodium-Water Reaction Test Loop (SWAT-2), and the Steam Generator Safety Test Facility (SWAT-3), at PNC in Japan. Particularly, experiments intended to investigate the failure propagation phenomena have been underway using SWAT-3 and data up to third or fourth failures have been obtained.

In addition, as related equipment, the leak detection system has been established for overall leak size, and this system includes in-sodium hydrogen meters, cover gas pressure gages, rupture disk burst detectors, and these in combination.

To date, a DBL for the MONJU steam generator has been selected as the leak rate equivalent to one plus three DEG failures, and the present paper describes the results of analysis to confirm the validity of the DBL using data obtained by sodium-water reaction studies so far. The analysis has been performed by a computer code, LEAP, which includes formulas based on the world sodium-water reaction test data. Initially, validation studies of the LEAP code have been carried out by comparing with the SWAT-3 experimental data mentioned above, and it has been confirmed that LEAP calculations estimate the extent of the failure propagation with enough conservatism. After these code verification studies, calculations for the MONJU system have started.

2. INITIATION OF LEAK AND MECHANISM OF ITS PROPAGATION

It is reasonable to assume that a very small or microscopic leak, rather than an intermediate or large leak, initiates the failure propagation. The initial leak might be caused by a micro-defect produced during fabrication processes or plant operations. This micro-leak would grow to a small or intermediate leak repeating a self-plugging/umplugging due to self wastage. Generally, it is difficult for the micro-leak to be detected with the present detection equipment, and therefore, corrective operations might not be expected. On the contrary, leaks larger than the micro-leak can be detected, but if automatic or manual operations were not taken adequately, the failure propagation, mainly due to wastage effects, produces the intermediate or large leak accident.

Table 1 shows the classification of the leak size with regard to their governing effects and detection methods. For each leak region, results of experimental studies and detailed discussions of their characteristics are presented in the following sections.

2.1 Initial Leak

A defect produced during the fabrication processes or the plant operations causes an initial leak. Referring to past accidents of sodium-heated steam generators, most of defects were found at welds (differing from the cases of fossil fueled steam boilers or light water reacter steam generators).[1]

The faulty weld may arise from the incompetent design or unskilled weld techniques. Since welds are considered as the most plausible leak potential source, careful attention must be paid to them. A size of defect produced during the fabrication depends on the quality control, especially post-weld inspection. As for the base metal, it will not be necessary to consider a initial defect if a careful quality control is applied.

On the other hand, a defect produced during the plant power operation might arise from pitting or stress corrosion cracking (SCC) due to the oxygen content in the water, fatigue due to the thermal cycle of the operation, or fretting due to the hydrodynamic vibration. However, the possibility of these defects are considered to be reduced reasonably by water and sodium quality control, the appropriate tube material specification, and the development of the tube support configuration. Moreover, the periodic ISI examinations would be able to detect flaws and prevent leaks prior to accidents.

2.2 Micro-leak

The leak jet of a micro-leak is not large enough to reach adjacent tubes; therefore, failure propagation does not occur, but their uncontinuous self-enlargement effects are of interest in this leak region. In the case of micro-leaks, there exists the possibility that the wall of the nozzle is corroded, and consequently a sudden increment of leak rate occurs when the inner diameter enlarges in all depth of the hole by self-wastage. Actually, leaks will often be detected at this time.

Significant analytical values of the self-enlargement phenomena are the multiplication factor of leak rate and the time required for enlargement. Although quantitative data for this leak region are not enough, data[2] reported by GE seem to be most reliable so far, thus the LEAP code adopts it with conservative modification taking into account of PNC's data.[3]

The enlargement time and the leak multiplication factor are expressed by following equations:

For $2\frac{1}{4}$Cr-1Mo steel;

$$T = \begin{cases} 4779.8 \quad G^{-0.5} & : \quad T_{Na} = 304 - 340\ ^\circ C \\ 377 \quad G^{-0.7} & : \quad T_{Na} = 469\ ^\circ C \end{cases} \qquad (2\text{-}1)$$

$$K = \begin{cases} 400 & : \quad G < 10^{-2} \text{ g/sec} \\ 30 & : \quad 10^{-2} \leq G < 10^{-1} \text{ g/sec} \\ 4 & : \quad G \geq 10^{-1} \text{ g/sec} \end{cases} \tag{2-2}$$

For austenitic stainless steel;

$$T = 150\ G^{-1} \tag{2-3}$$

$$K = 4 \tag{2-4}$$

Where, T is the enlargement time in seconds, K is the leak rate multiplication factor, G is the leak rate in g/sec and T_{Na} is the sodium temperature in centigrade.

2.3 Small Leak

The small leak region described here is defined as the leak region in which a tube adjacent to a leak site is wasted by such a leak jet. Many quantitative formulations to estimate their effects are available from experimental and analytical research reported in the world, since the first wastage failure was found in the Fermi reactor plant.

There are many factors which influence the wastage rate, and they can be summarized in the following three categories:

1) Water/steam conditions and leak jet configuration.
2) Sodium conditions.
3) Target tube conditions.

These factors can be represented by such as the target tube material, the distance between leak and target, the leak rate, and the sodium temperature. These parameters are important for the formulation of the wastage rate which is utilized in the computer analysis of the LEAP code.

The formulas of the wastage rate in the small leak region used in the LEAP code are derived from data obtained by using the SWAT-2 test loop in Japan.[4,5)]

$$W_R = \frac{4410}{L^*} \exp\left[- \left\{ 0.255 \left(\ln \frac{G}{5.12} \right)^2 + \frac{5460}{T_{Na}} \right\} \right] \tag{2-5}$$

$$L^* = \begin{cases} L & : \quad L \geq 30\ D \\ 30\ D & : \quad L < 30\ D \end{cases}$$

For austenitic stainless steel;

$$W_R = \frac{9205}{L^*} \exp\left[- \left\{ 0.287 (\ln \frac{G}{3.19})^2 + \frac{7180}{T_{Na}} \right\} \right] \qquad (2\text{-}6)$$

Where, W_R is the wastage rate in mm/sec, which is defined as the depth of wastage per unit time, L is the distance between leak and target in mm, and D is the leak hole diameter in mm. The tendency is that austenitic stainless steel is superior to $2\frac{1}{4}$Cr-1Mo steel concerning the wastage, and the maximum wastage rate appears when the leak rate is about 5 g/sec for $2\frac{1}{4}$Cr-1Mo steel.

2.4 Intermediate Leak

In the intermediate leak region, wastage is the predominant mechanism for tube failures as well as in the small leak region. While only one tube is wasted in the latter region, several tubes are exposed in a jet and wasted simultaneously in the former region. The border between these two leak regions is drawn at the leak rate of 10 g/sec in the MONJU configuration.

Intermediate leak wastage is believed to provide two main effects on the target material, that is, one is an erosion effect by collision of the leak jet in the close vicinity of the leak point, and the other is an corrosion effect by sodium-water reaction products slightly appart from the leak site.

Besides wastage, overheating is another potentially significant mechanism for secondary failures, when the leak rate seems to be rather large in the intermediate leak region. Overheating is a phenomenon whereby mechanical strength of tubes decreases due to a transient heatup by the hot reaction products from a leak to the temperature where overheated tubes cannot contain the pressurized water. Especially, the term of overheating used in the present paper means the mechanism that tubes burst by the reaction heat rapidly enough not to have thinned by wastage. Consequently, overheating is considered more serious than wastage because the times for tube failures will be as short as a few seconds, and the failure size will be large, up to one DEG (if indeed such failures occur).

Intermediate leak test series have been conducted using the Large Leak Sodium-Water Reaction Test Rig (SWAT-1) at O-arai Engineering Center in PNC.[6,7] The primary objectives of these tests are to obtain data on the time, size and mechanism for secondary failures in the intermediate leak region, and to arrange them as empirical formulas so as to be utilized in the LEAP code. However, the arranging work is difficult in the intermediate leak region because there is a hydraulic complexity including reflection and scattering of the jet in the tube bundle, besides the complexity of the chemical reactions in the small leak region. The formulas to be utilized in the LEAP code have been derived from the data from not only the intermediate leak tests in SWAT-1 but the large leak and failure propagation tests in SWAT-3 in order to be applicable to both intermediate and large leaks. These will be explained in paragraph 2.6.

Characteristics of intermediate leak wastage are, for example, very weak dependence of the sodium bulk temperature and the jet collision angle which is defined to be between the jet direction and the tangent of the surface of a target tube on the wastage rate. As for multi-wastage phenomena, in the case that the water leak rate is more than 100 g/sec, several tubes are simultaneously damaged at the wastage rate of 0.05 to 0.07 mm/sec that is the maximum wastage rate obtained in the intermediate leak wastage tests.

2.5 Large Leak

Large leaks are generally considered to be the leak region where acoustic pressure waves, differential thermal expansion or fluid drag forces resulting from the primary leak should be the dominant secondary failure potential rather than wastage. The lower border of this region is generally drawn at the leak rate of 2 kg/sec. Seven tests, Run-1 through Run-7, in SWAT-3 were conducted in order to certify the integrity of the MONJU steam generators against the large leak sodium-water reaction accidents, and the test results have already been reported in another paper.[8)]

These tests concluded that heat transfer tubes as well as the pressure boundaries of steam generators are firm for large leaks. That is, no failure propagation occurred. Especially, in Run-1 through Run-6 where the water was injected in the helically coiled region, neither bowing, bulging nor thinning of tubes was observed. While, in Run-7 where the water was injected in the down-comer region, some tubes were bowed and the maximum wastage rate of 0.003 mm/sec was obtained for the first time in the large leak region. This value is taken into account in the expressions in the LEAP code as described paragraph 2.6.

As for overheating, there was no trace suggesting such an occurrence in the large leak tests of SWAT-3 during the injection of about ten seconds, which seems to be long enough to assure the possibility of overheating. The reason why the secondary failure potential from both overheating and wastage is very low in the large leak region is that due to the voiding action caused by the produced hydrogen, the stationary flame front cannot exist.

2.6 Failure Propagation Tests

Failure propagation tests[7,9] (Run-8 through Run-14 at present) have been carried out using SWAT-3 since 1978 in order to provide data for selecting the DBL of the MONJU steam generators, especially data on time, size, and mechanism of failure propagation in all leak range.

In these tests, some data were obtained in the range of the leak rate about 0.5 to 2.0 kg/sec which is in the middle range between the intermediate leak and the large leak. Main test results are as follows:

1. The wastage rate was slightly lower than that in the intermediate leaks.
2. As for failure mechanism, several tubes filled with pressurized nitorgen gas burst.

However, strictly speaking, the latter type of failure is different from overheating mentioned previously. Since it took about a minute for these tubes to burst after the initiation of water leak, they seem to have been thinned significantly by wastage. And it is important that for the tubes filled with pressurized water the burst type failure did not occur.

Expressions described below are derived from these data as well as the intermediate leak wastage data in SWAT-1 and the large leak data in SWAT-3.

As for the wastage rate, W_R, in mm/sec, an expression as below is used in the LEAP code.

$$W_R = \begin{cases} A & L < 50 \text{ mm} \\ A \exp\left\{-B\ (L-50)^2\right\} & L \geq 50 \text{ mm} \end{cases} \qquad (2\text{-}7)$$

$$A = \begin{cases} 7 \times 10^{-2} & 10 \leqq G < 800 \text{ g/sec} \\ 244\ G^{-1.22} & G \geqq 800 \text{ g/sec} \end{cases}$$

$$B = \begin{cases} 8.33 \times 10^{-4}\ G^{-0.4} & 10 \leqq G < 200 \text{ g/sec} \\ 1.00 \times 10^{-4} & 200 \leqq G < 500 \text{ g/sec} \\ 5.48 \times 10^{-11}\ G^{2.32} & 500 \leqq G < 1000 \text{ g/sec} \\ 5 \times 10^{-4} & G \geqq 1000 \text{ g/sec} \end{cases}$$

G and L mean the water leak rate and the distance between the leak nozzle and a target tube, respectively. Fig. 1 shows the relation between A the maximum wastage rate and G.

As for the failure size, it depends on the type of failures such as pit type, toroidal type and burst type as shown in Fig. 2. The diameters of failure holes in the case of pit and burst type are expressed as a function of the diameter of initial leak hole like this,

$$D_2 = 4.5\ D_1^{1.25} \tag{2-8}$$

The maximum of the diameter D_2 is limited to that equivalent to one DEG in the case of failure in the upper tube bundle region in evaporators. However, it is limited to 5.7 mm diameter, the largest secondary failure observed in the water-filled tubes, in the lower region in evaporator. The test results showed that water-filled tubes did not burst in SWAT-3 test.

The diameters of the failure hole in the case of toroidal type are expressed as below:

$$D_2 = 1.8\ D_1^{1.5}\ L^{0.5} \tag{2-9}$$

The expression above is only applied in the case that a flame circle with a diameter of D_2 is perfectly projected on the surface of one tube.

In the sodium-water reaction tests of intermediate and large leaks in PNC, only $2\frac{1}{4}$Cr-1Mo steel is used as the tube material, and no data have been obtained for stainless steel so far. Therefore, the wastage rate for the stainless steel tubes which are planned to be used in superheaters is derived from the proportional relation between the data of two materials in the small leak tests.

3. OUTLINE OF THE LEAP CODE

LEAP (Leak Enlargement And Propagation) is a computer code which was developed in order to numerically simulate the failure propagation process from the occurrence of initial water leak, to the leak detection, to the water dump, and to the termination of the reaction, and to calculate the maximum leak size due to the failure propagation.[1)]

3.1 Modeling

The flow diagram of the failure propagation phenomena is outlined in Fig. 3. Then the analysis model consists of elements as below:

1. Operational conditions of the steam generator.
2. Leak data (leak rate, jet direction, jet length, etc).
3. Selection of the target tubes.
4. Calculation of target wastage.
5. Tube penetration.
6. Self-enlargement of the leak nozzle.
7. Leak detection.

These elements are linked each other in the LEAP code as shown in Fig. 4.

4. ANALYTICAL RESULTS BY LEAP

The analyses of the failure propagation phenemena have started using the LEAP code, and the work consists of two stages: the first one is to validate this code using the data of full term simulation tests in SWAT-3, and the second is application to the safety analyses of the MONJU steam generators.

4.1 Code Verification by SWAT-3 Data

Run-14 in SWAT-3 is the full term simulation test; that is, all of fifty-six tubes were filled with the subcooled water of 150 ata (15 MPa) pressure, and the water injection (whose initial value was 20 g/sec) was stopped after the decrease of water supply tank pressure which simulated the water dump at the large leak sodium-water reaction accidents in the real steam generator plant system.

Fig. 5 shows the outline of the failure propagation process in

Run-14 of SWAT-3. The initial leak tube was No.1 and the direction of the jet was 11 degrees counter clockwise from the bottom. On the other hand, Fig. 6 shows the LEAP calculation result of the failure propagation process on the same conditions as Run-14 of SWAT-3. From this comparison, it seems that the difference between the test data and the calculation increases with the progress of propagations. This is because the empirical formulas used in the LEAP code have been always derived conservatively.

A comparison between the time history of the Run-14 test result and that of LEAP calculation is shown in Fig. 7. As shown plainly, the calculation always provides a conservative value for both the time and the size for the failures. The maximum water leak rates of the test result and the calculation are 1.6 kg/sec and 16 kg/sec, respectively.

From the comparisons above, it is concluded that the present LEAP code cannot exactly identify the failed tube and the jet direction after a few stages of propagation. However, it can conservatively predict the total leak size and the times of individual tube failures for the MONJU analysis. This code validation work is ongoing and it will use the data of simulation tests in SWAT-3 which are planned to be conducted in the near future.

4.2 Application of LEAP to MONJU Analyses

An objective in applying the LEAP code to MONJU analyses is to confirm the DBL of the present design, including the related equipment. Parameters are selected as indicated below:

∘ Steam generator	<u>evaporator</u> or superheater
∘ Leak site	<u>helical region</u> or down-comer region, and <u>upper</u> or <u>lower region</u>
∘ Operational conditions	<u>full load</u> or partial load
∘ Leak detection	in-sodium hydrogen meter, in-cover gas pressure transducer, or <u>rupture disc burst signal</u>
∘ Water blowdown	<u>emergency dump</u> or ordinal dump

The reference case indicated by the underlines above is regarded to be most serious for the plant as a whole. For example, it is postulated that both the hydrogen meter and the pressure transducer in cover gas have no credit and only a rupture disc burst signal is used as the leak detector. Hence, a pressure transient calculation in the cover gas line is involved in this code. And the water header pressure decrease during the blowdown is derived from the RELAP code with sufficient conservatism, that is, it adopts 120 sec as the period of blowdown when the pressure decreases from about 150 ata (15 MPa) to 8 ata (0.8 MPa).

The same tube configuration as Run-14 in SWAT-3 is used in these calculations, however, the total number of tubes increases to 144 added to surround the tubes of Run-14. The jet direction of the initial leak is also chosen to be the same one as in Run-14 because it was confirmed to be sufficiently conservative in the sensitivity analysis.

Fig. 8(a) and 8(b) show the examples of failure propagation analyses for the MONJU steam generators by the LEAP code in the case that initial water leak rates G's are 0.1, 1.0, 10 and 1,000 g/sec. In the first and second cases, it takes hours for the initial leak to grow by self-enlargement. In the third and fourth cases, the water leak rates reach their peak value in a few minutes after several steps of propagation.

The relations between the initial leak rate and the maximum leak rate are shown in Fig. 9 and Fig. 10 in the cases that the leak sites were in the lower and upper helical-coil region in the evaporator, respectively. Each circle means a calculation case. In the lower cases most of the values of maximum leak rate are in the range of one or two DEG's, though there is some fluctuation. The maximum value is 27 kg/sec; that is slightly larger than two DEG's. On the other hand, in the upper cases, the maximum is 37 kg/sec, which is slightly lower than three DEG's. In both cases, there is a tendency that the initial leaks in the small leak region provide the larger maximum values and that the leak rates do not exceed the initial value in the large leak region. The reason why the leak in the upper site provides the larger maximum value than that in the lower site seems to be due to the difference of the upper limit of the secondary failure size shown in Fig. 2.

5. CONCLUSION

Safety analyses for MONJU steam generators have been performed using the LEAP code which involves the world investigation data on the sodium water reaction. The comparison with the SWAT-3 failure propagation tests confirmed that the LEAP code can consevatively and validly presume the total leak size. As the results of the application to MONJU analyses, the one plus three tube DBL selected formerly has been ascertained to be valid in the case of the helical-coil region in the evaporator. Other calculation cases such as for the down-comer region, the superheater, and considering other detection systems, are underway now.

6. REFERENCES

1. Sub-Committee on the Investigation of Large Sodium-heated Steam Generator, "Studies on the Water Leaks and Safety Devices of Large Sodium-heated Steam Generators (3)", PNC SJ 254 80-01, March, 1980.
2. J.A. Gudahl and P.M. Magee, "Micro Leak Wastage Test Results", GEFR-00352, March, 1978.
3. M. Kuroha, et al., "Preliminary Study of Micro-defect Self-wastage on $2\frac{1}{4}$Cr-1Mo Steel Nozzles for LMFBR Steam Generators", PNC SN941 80-135, June, 1980.
4. Y. Daigo, et al., "Evaluation of Wastage due to Steam Leaks", DOE/PNC Plant Components W/Gr First Joint Steam Generator Seminar, PNC SA013 KWG 78-01, February, 1978.
5. N. Kanegae, et al., "Wastage and self-Wastage Phenomena Resulting from Small Leak Sodium-Water Reaction - Studies of Small Leak Sodium-Water Reactions (7) -", PNC SN941 76-27, March, 1976.
6. H. Tanabe, et al., "Intermediate Leak Wastage Test of Heat Transfer Tube of LMFBR's Steam Generator", PNC SN941 80-27, February, 1980.
7. H. Tanabe, et al., "Experimental Studies on Leak Propagation of Heat Transfer Tubes by SWAT-1 and SWAT-3", DOE/PNC Plant Components W/Gr Second Joint LMFBR Steam Generator Seminar, PNC SA013 KWG81-03, June, 1981.
8. H. Hiroi, et al., "Results and Conclusions from PNC's Large Leak Tests", DOE/PNC Plant Component W/Gr Second Joint LMFBR Steam Generator Seminar, PNC, SA013 KWG81-03, June, 1981.
9. H. Tanabe, et al., "Results of Failure Propagation Tests in Steam Generator Safety Test Facility (SWAT-3) - Report No.1", PNC SN941 81-05, January, 1981.

Table 1 Definition of Leak Size

Leak Region	Micro Leak	Small Leak	Intermediate Leak	Large Leak
Definition (Roles)	• non damage leak rate • possibility of self-wastage	• one-tube (direct) wastage	• multi-tube wastage	• hydrodynamic and temperature increment effects
Detection	• undetectable (even an identification of a faulted steam generator is difficult)	• increment of hydrogen concentration	• increment of hydrogen concentration • increment of pressure • fluctuation of sodium level	• increment of pressure • burst signal of rupture disk
Extent of Failure	• adjacent tube is not damaged	• one tube is damaged by wastage.	• more than two tubes are damaged by wastage. The extent is not larger than four DEG.	• large scale rupture; the system integrity is retained by action of the pressure relief system

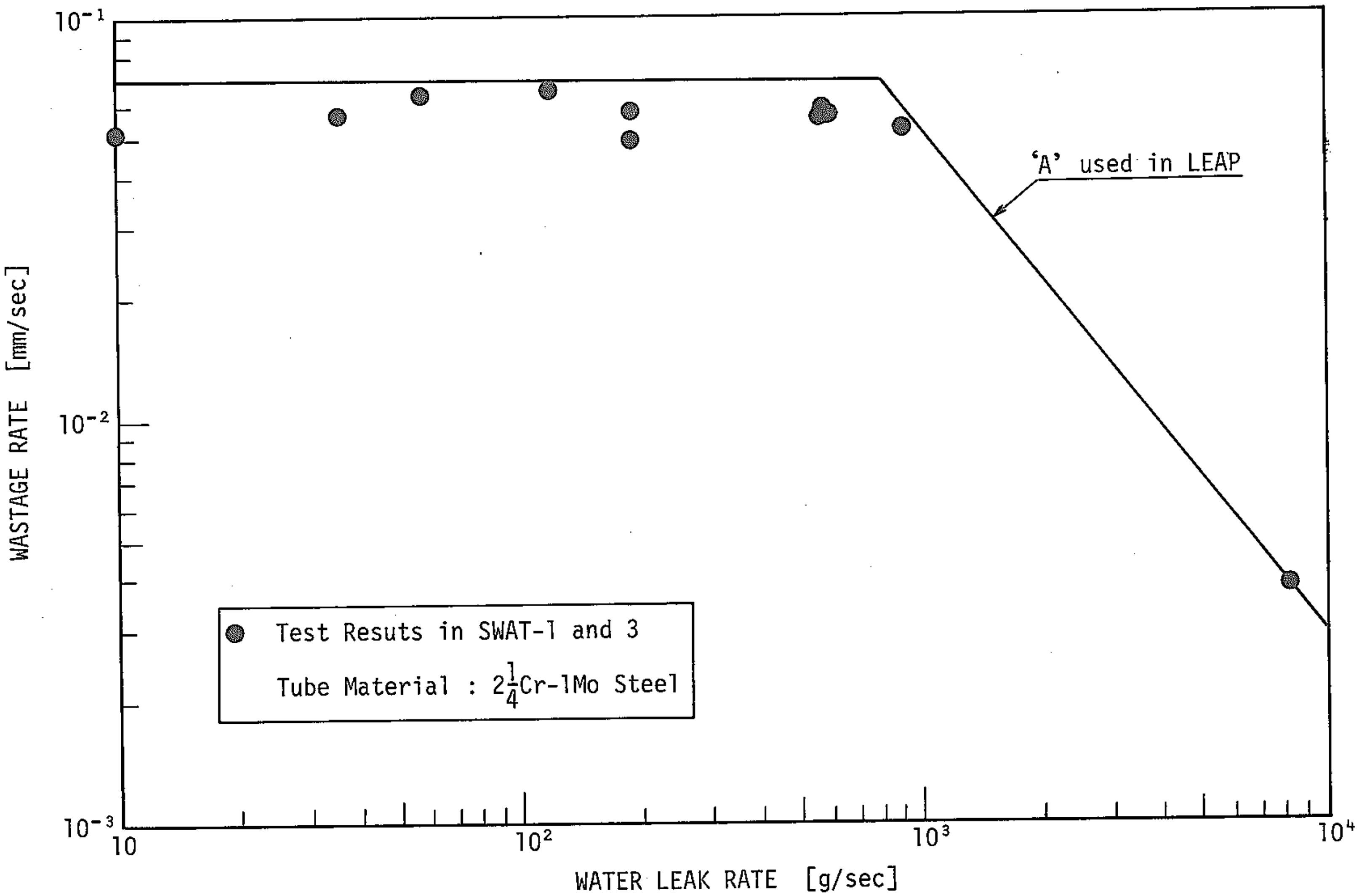


Fig. 1 Relation between Water Leak Rate and Tube Wastage Rate

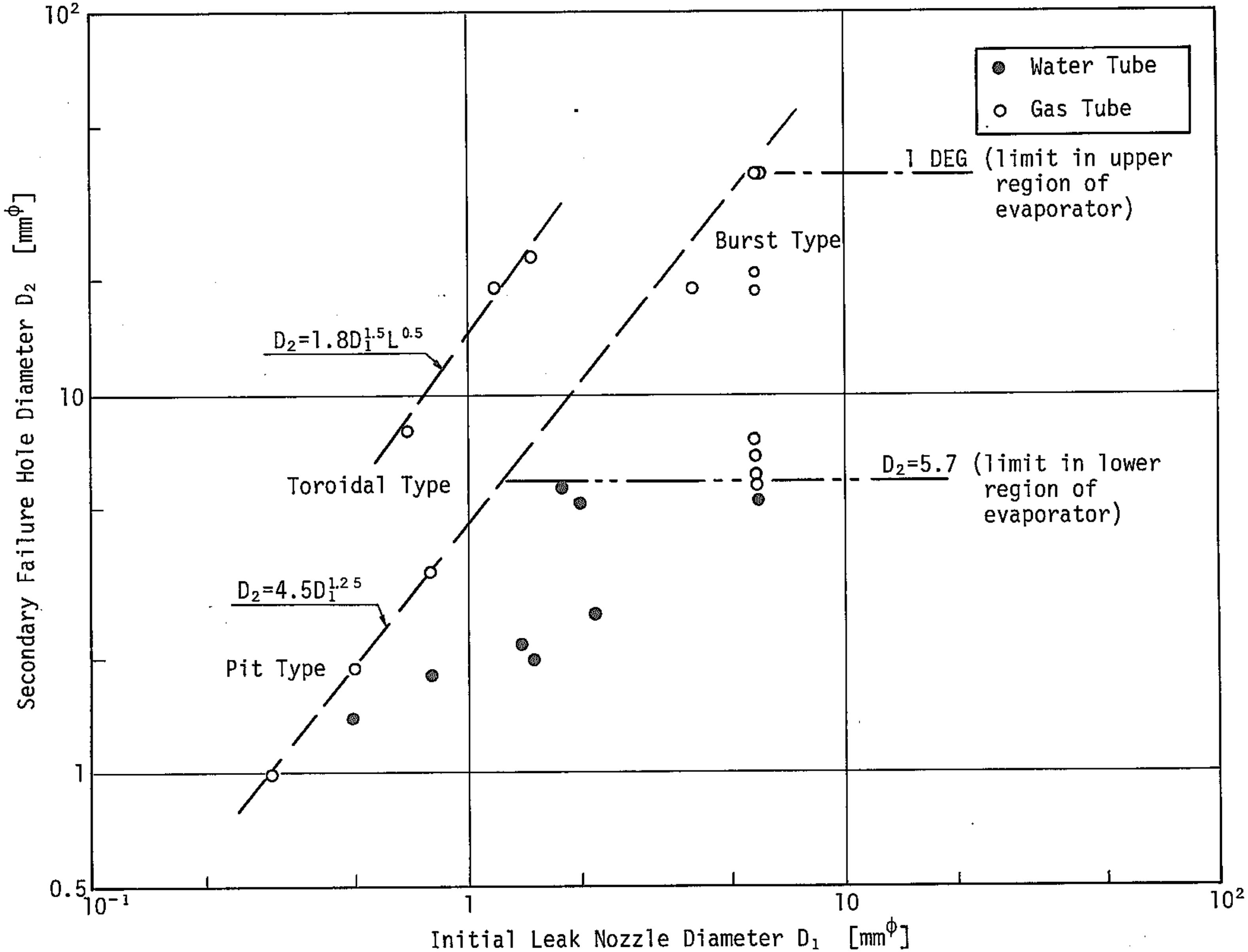


Fig. 2 Relation between Initial and Secondary Leak Diameter

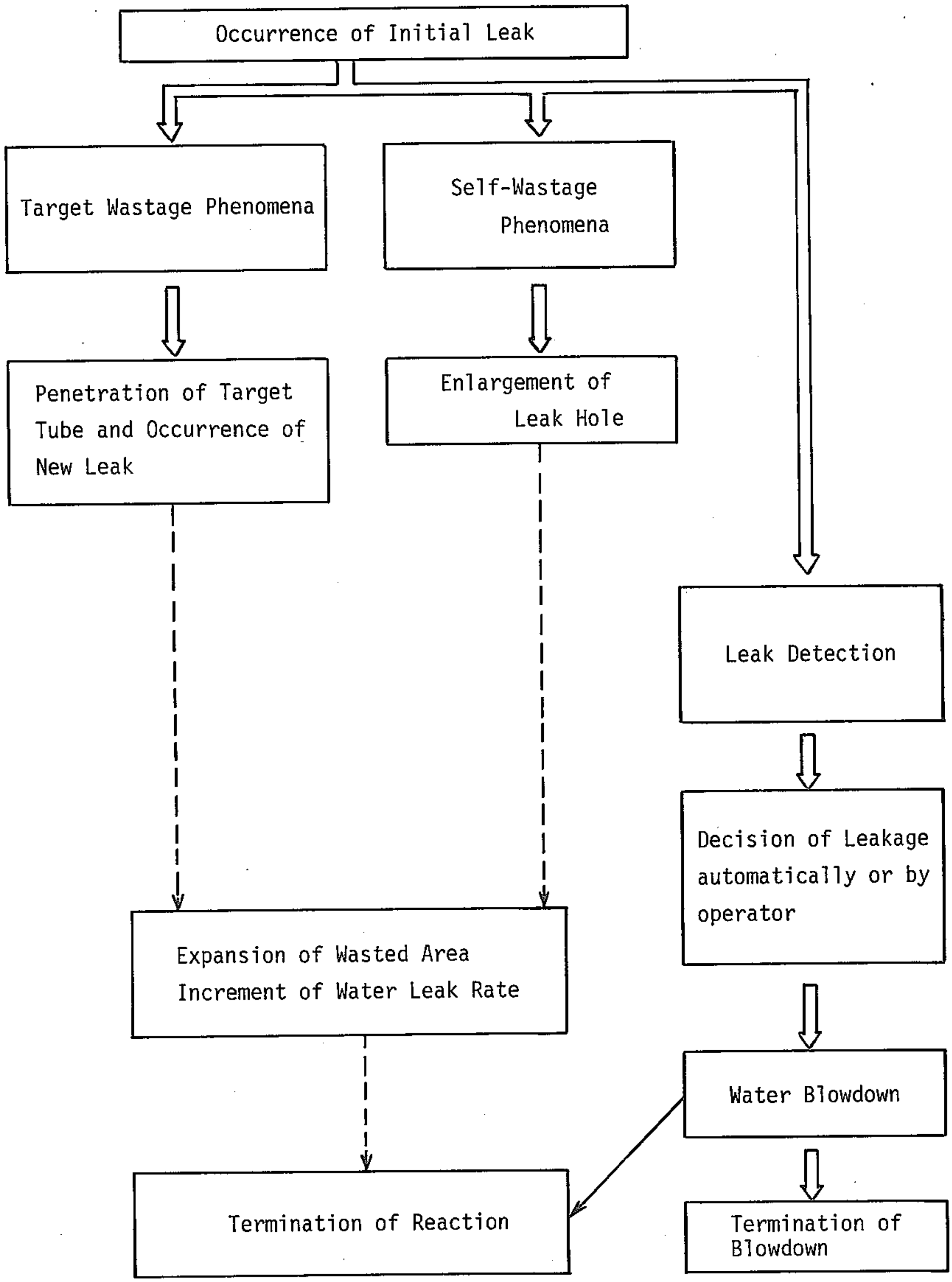


Fig. 3 Modeling of Steam Generator Failure Propagation Phenomena

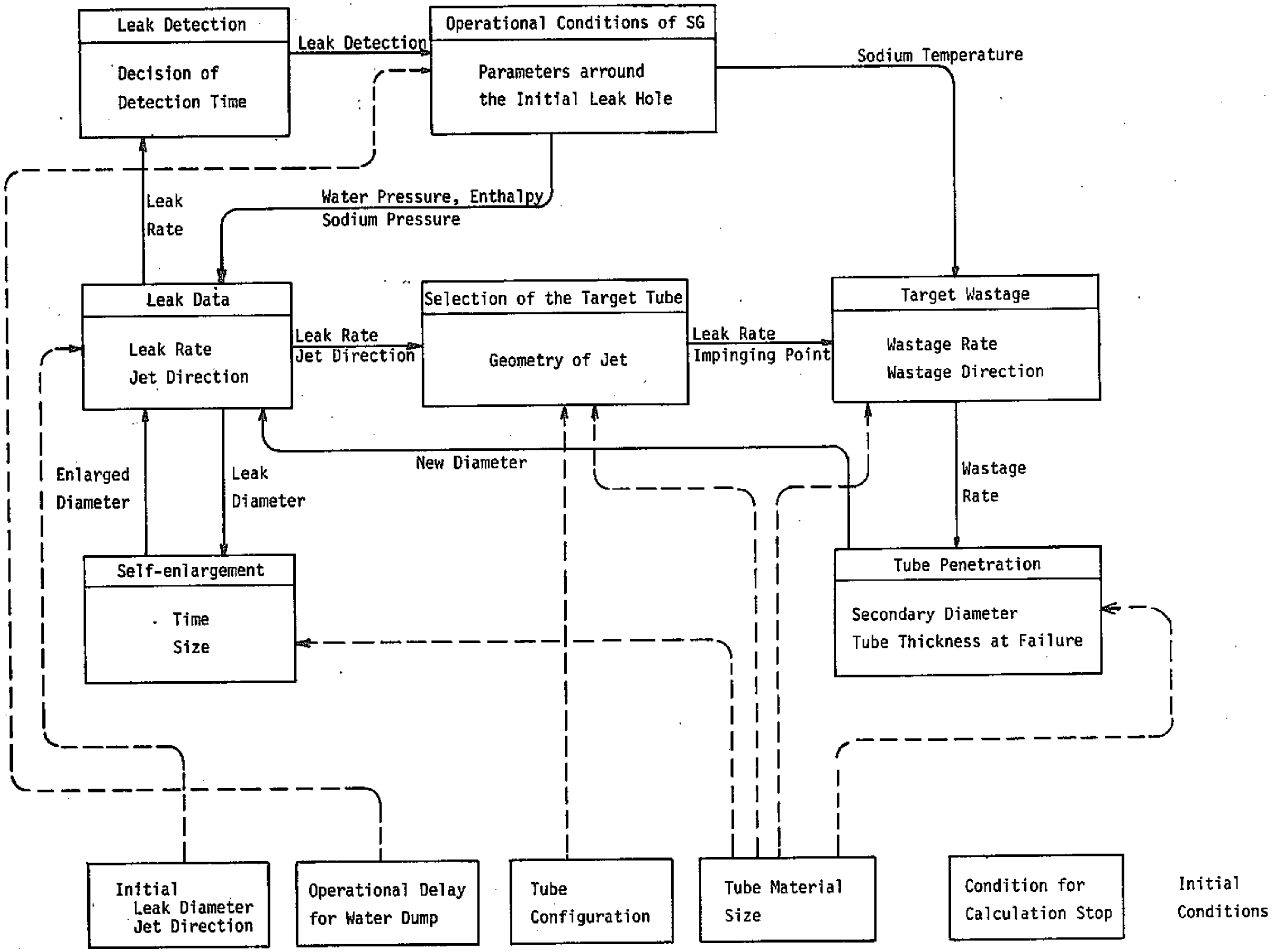


Fig. 4 Structure of LEAP Code

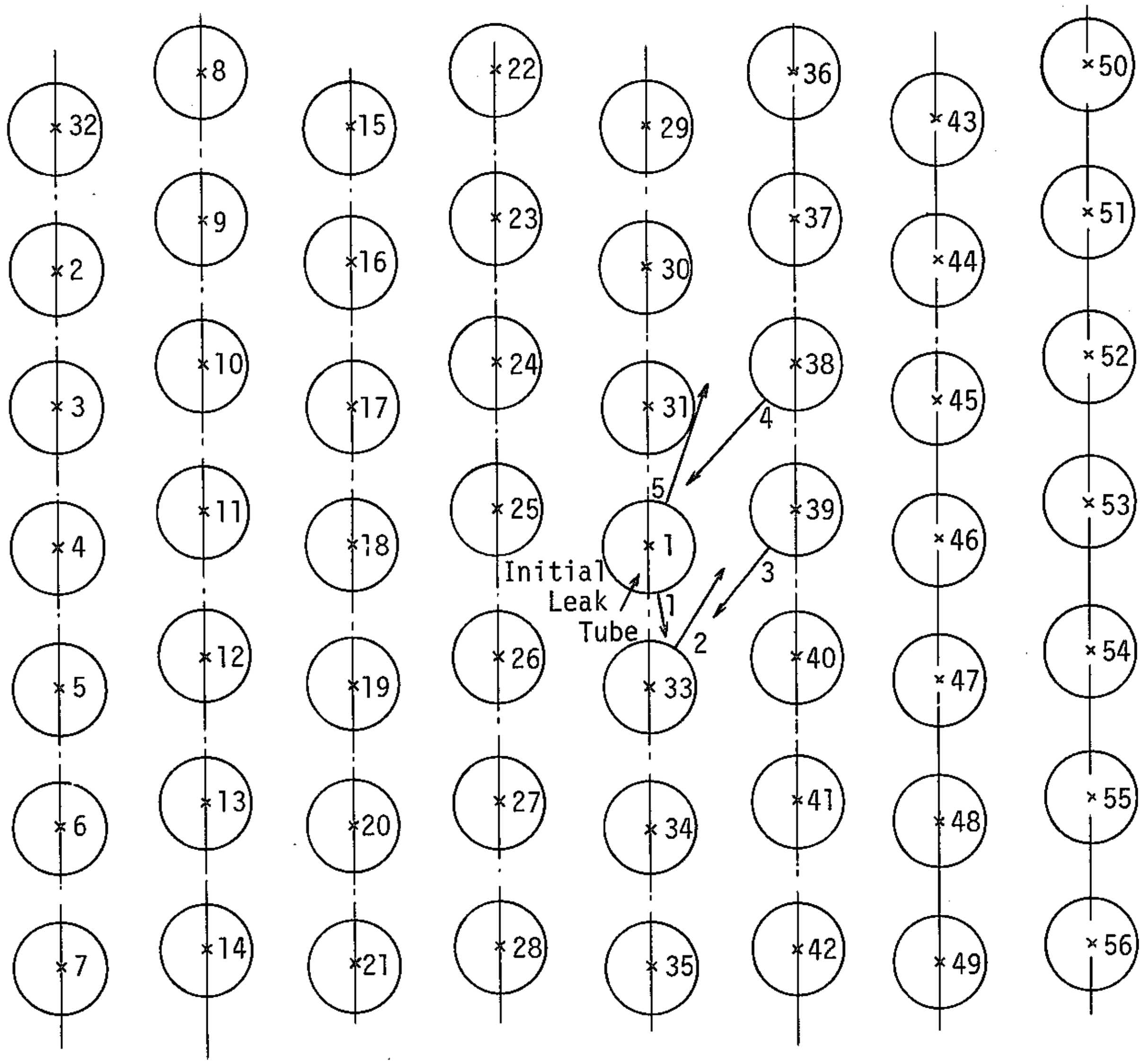


Fig. 5 Failure Propagation Test Results in Run-14, SWAT-3

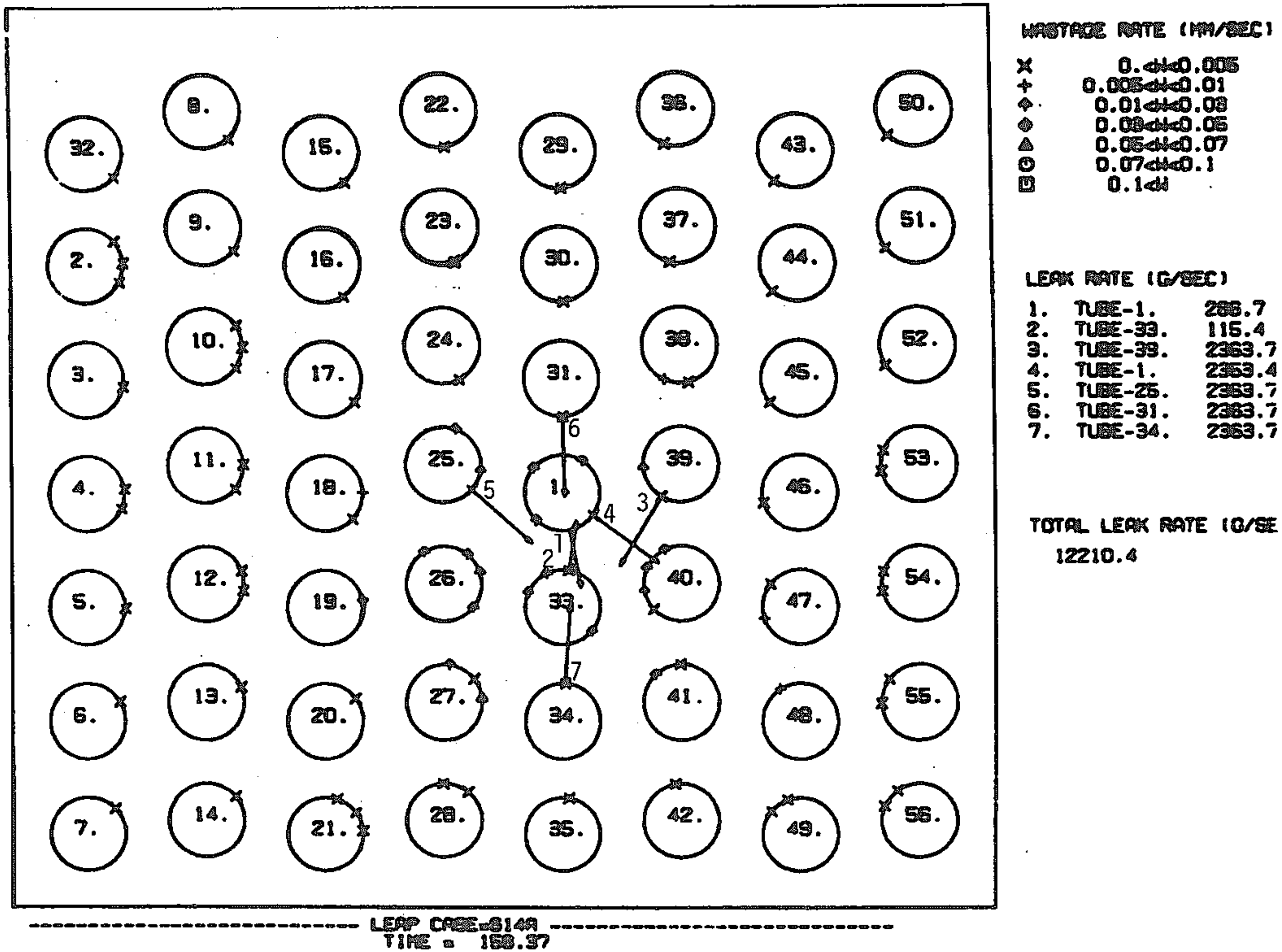


Fig. 6 LEAP Calculation Result of Failure Propagation Progress

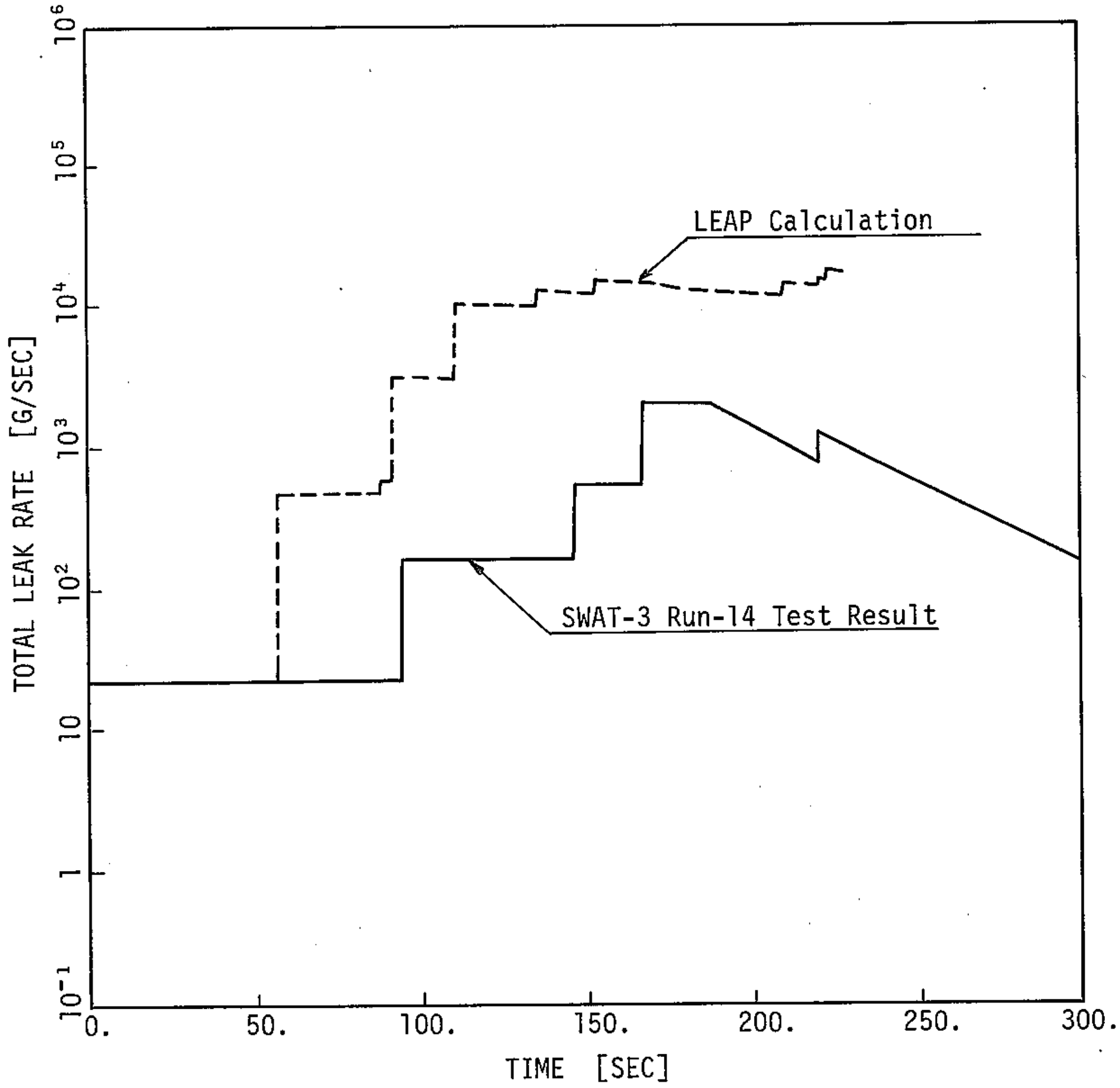
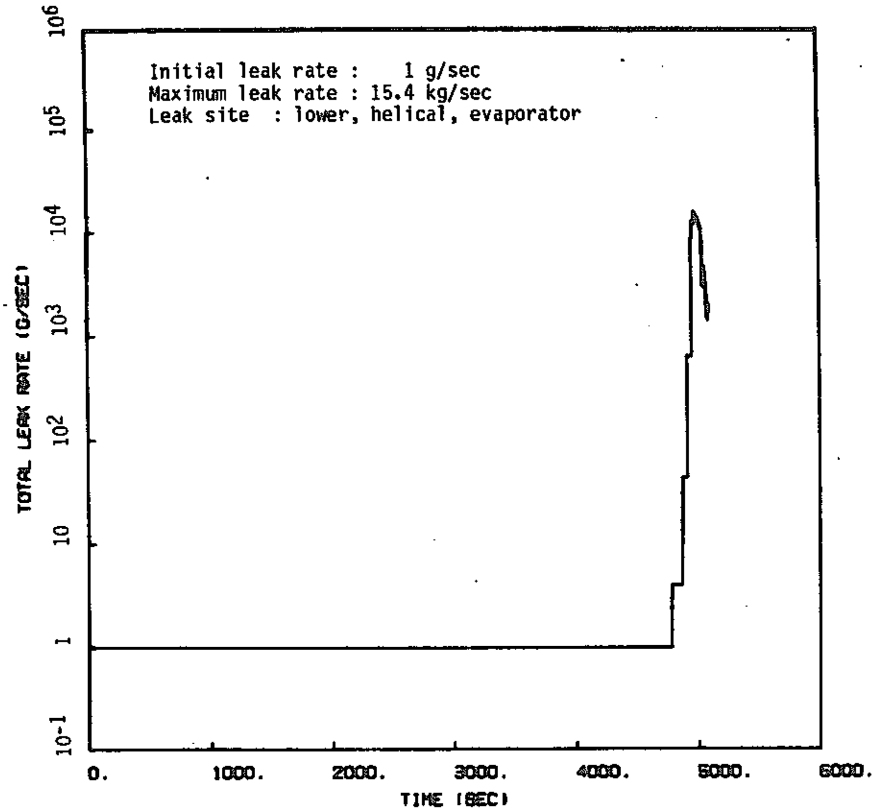


Fig. 7 Comparison of Time History between SWAT-3 Test Data and LEAP Calculation



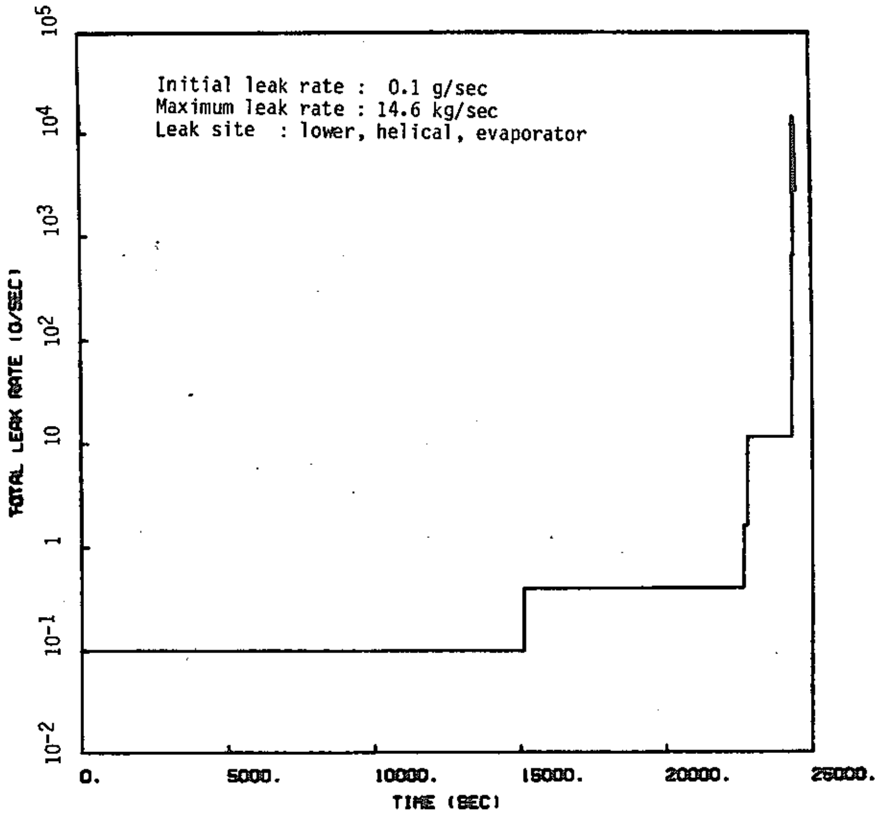
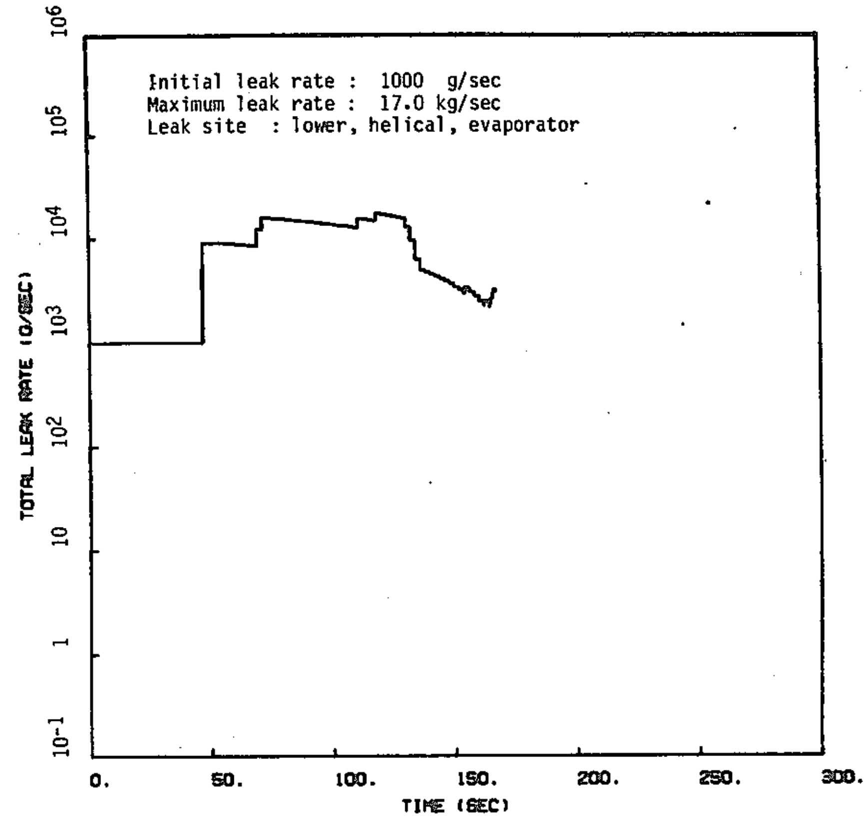


Fig. 8(a) LEAP Calculation Results for MONJU Steam Generator



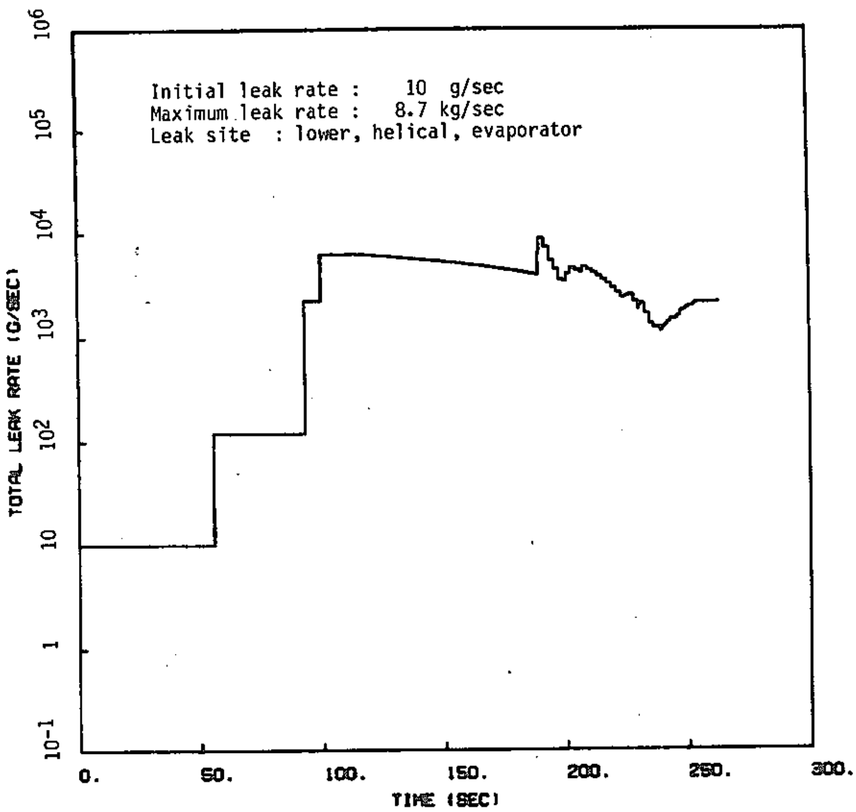


Fig. 8(b) LEAP Calculation Results for MONJU Steam Generator

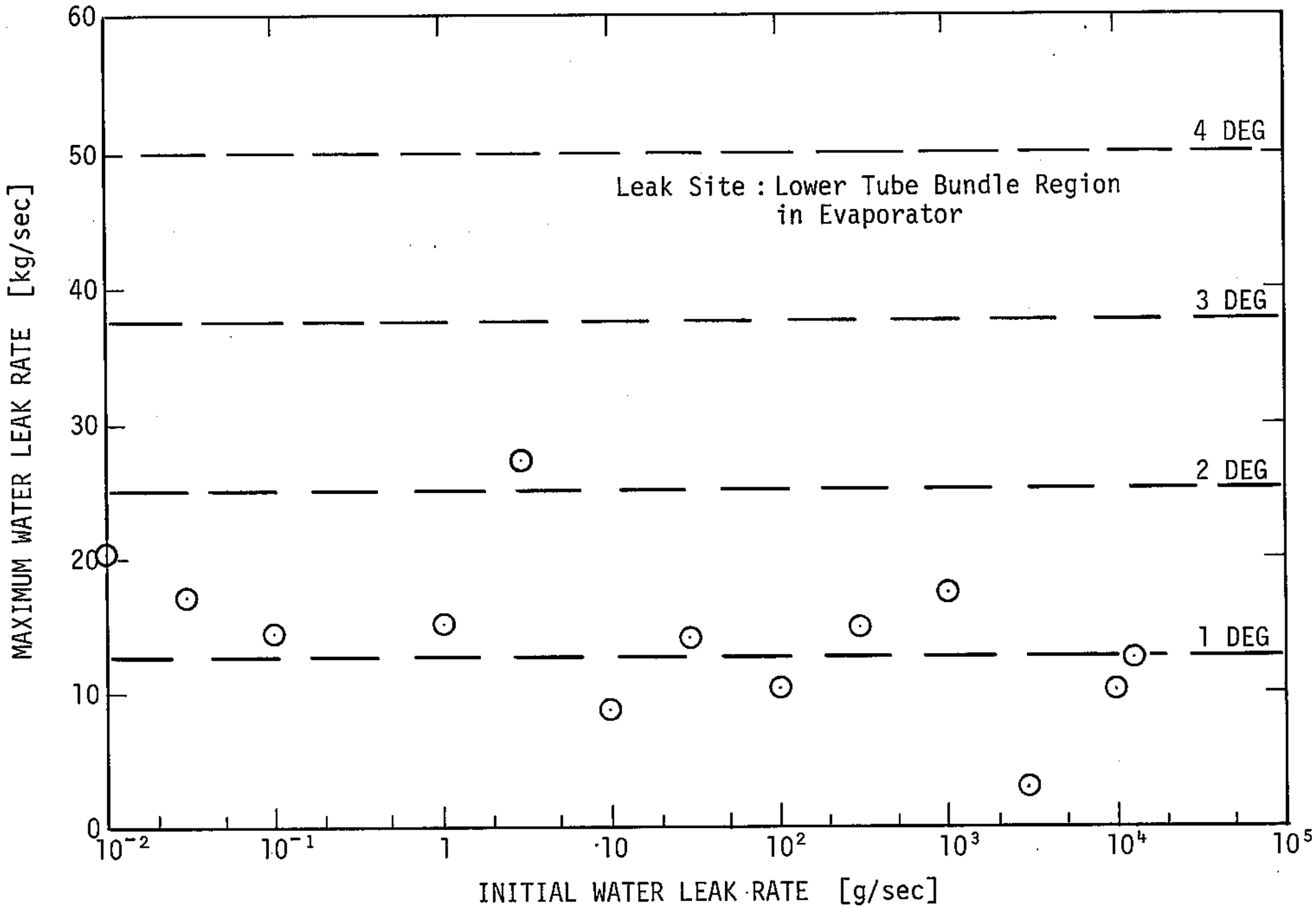


Fig. 9 Maximum Water Leak Rate for Lower Region Leak

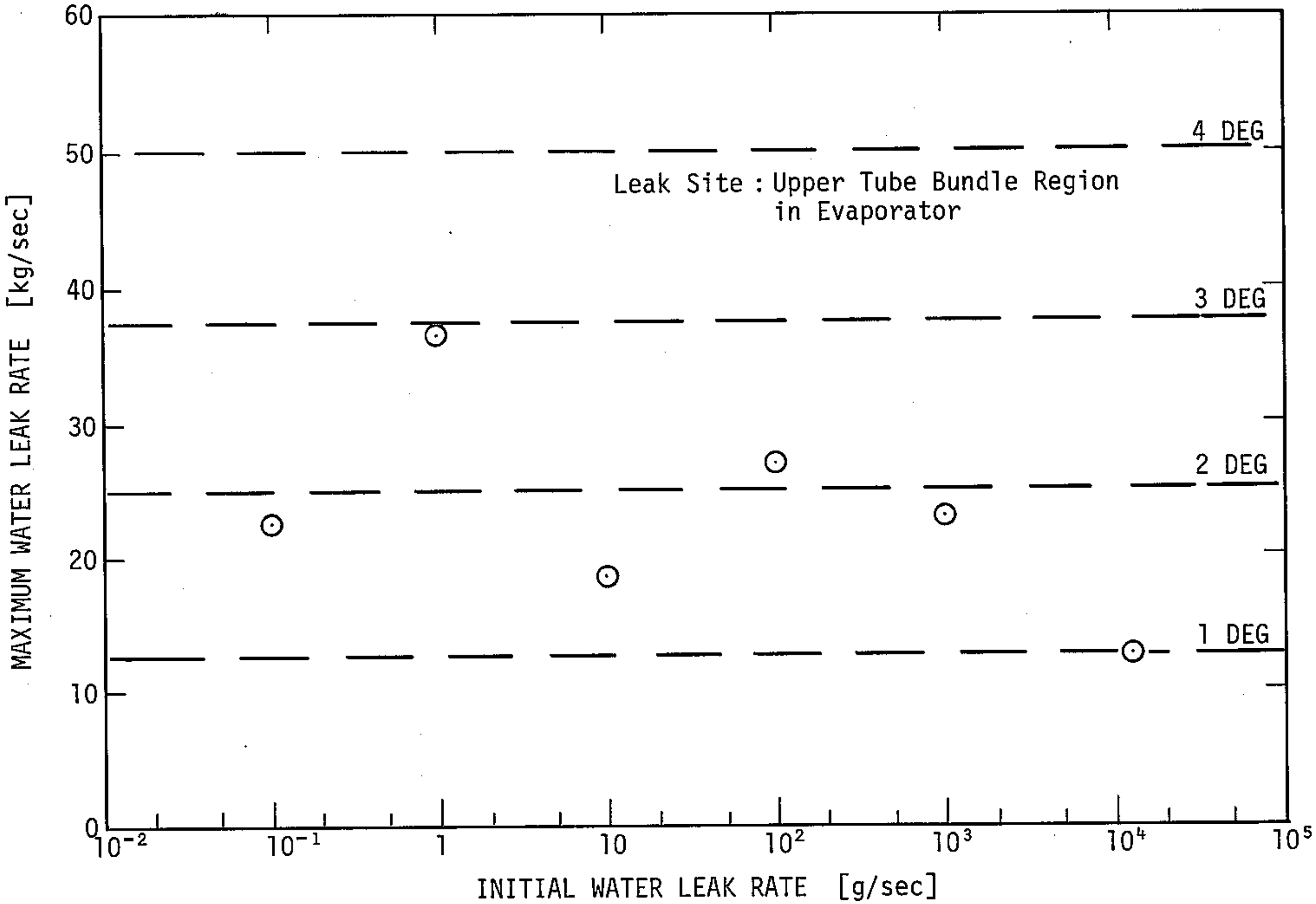


Fig. 10 Maximum Water Leak Rate for Upper Region Leak