UNITED STATES
NUCLEAR REGULATORY COMMISSION
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June 7, 1996

MEMORANDUM TO: Ashok C. Thadani, Associate Director
for Inspection and Technical Assessment
Office of Nuclear Reactor Regulation

Brian K. Grimes, Acting Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

FROM: M. Wayne Hodges, Director M. Wayne Hodges
Division of Systems Technology
Office of Nuclear Regulatory Research

SUBJECT: REASSESSMENT OF THE ASSUMPTIONS AND PROPOSED ALTERNATIVE METHOD FOR DETERMINING RADIOLOGICAL CONSEQUENCES OF MAIN STEAM LINE BREAK AND STEAM GENERATOR TUBE RUPTURE

In a memorandum dated August 18, 1995, it was agreed that, in support of the rulemaking effort on steam generator tube integrity, RES would reassess the assumptions for determining the radiological consequences of a main steam line break accident. We have completed this reassessment which is provided in Attachment A. Our reassessment addresses seven issues related to the dose analysis and is primarily concerned with treatment of the iodine release. As discussed in Attachment A, we believe that the current assumptions are bounding and that there are areas where these assumptions could be relaxed.

We also agreed to pursue alternative methods for radiological consequence assessment for main steam line break conditions. We have developed an alternative method which utilizes Monte Carlo sampling, similar to that suggested by Dr. Dana Powers (ACRS) in his memorandum of August 17, 1994, and developed by EPRI, in draft technical report TR-103878 of March 1994. This alternative method is described in Attachment B and involves replacing the current approach of bounding assumptions for each of several individual elements with an integrated probabilistic treatment. Through a Monte Carlo analysis, a probability distribution of dose can be constructed. The probability distribution of dose can then be used as the basis for licensing in lieu of the deterministic bounding calculation. We have performed a radiological consequence assessment with this alternative method. This assessment used the EPRI-proposed spiking magnitude distribution for PWRs and the dispersion factor distribution for Site A of WASH-1400. This assessment also uses as a possible acceptance criterion, that the probability of exceeding 300 rem thyroid be less than .01.

This assessment is discussed in Attachment B. Use of the new methodology could result in an increased allowable leak rate greater than the current allowable leak rate while still providing an adequate margin in the safety assessment.

Please contact Jason Schaperow (415-5907) or Charles Tinkler (415-6770) of my staff if you should have any questions.

Attachment: As stated

cc: CMiller TReed
REmch RJones
DLurie

Distribution: Schaperow, Tinkler, Ader, King, Hodges, Murphy, Speis, Morrison, Soffer, Mayfield, Shao, Ridgely

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ATTACHMENT A

Reassessment of the Assumptions for Determining Radiological Consequences of Main Steam Line Break and Steam Generator Tube Rupture

In an August 18, 1995, memorandum from RES (W. Hodges) to NRR (A. Thadani), it was agreed that RES would reassess the assumptions used for determining the radiological consequences associated with a main steam line break (MSLB). This reassessment addressed the iodine spike release rate associated with a MSLB and also addressed the following issues for a SGTR: (1) iodine spiking model, (2) history of iodine spike concentrations, (3) release rate for accident-initiated iodine spike, (4) partition coefficient for iodine, (5) plateout in the steam generator, steam lines, ADVs, or MSSVs, and (6) iodine species.

The assessment of radiological consequences for MSLB and SGTR accidents currently consists of a dose calculation using conservative assumptions for the following three locations: exclusion area boundary, low population zone distance, and control room. The dose calculation can be broken down into five elements characterizing the release from the fuel to the eventual absorbed dose; (1) release from fuel into primary coolant, (2) transport of primary coolant to the secondary side of the steam generator, (3) transport through the secondary side of the steam generator and secondary piping/ADV/MSSV to the environment, (4) atmospheric dispersion, and (5) absorbed dose to the individual. The Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP) methodology is the same for both the SGTR and MSLB. However, in implementing the SRP methodology, there has been a difference in the treatment of scrubbing and dilution in the steam generator; credit is given for these processes in a SGTR but not for a MSLB. Credit is given in a SGTR, because the steam generator tubes are covered with water during this accident. Credit is not given for a MSLB, because the steam generator is quickly emptied as a result of the break and subsequent loss of feedwater.

Iodine Spiking:

Pre-SRP Work:

In the early 1970's, the Atomic Energy Commission and the nuclear industry became aware of the iodine spiking phenomenon, that is, iodine concentration in the reactor coolant increased up to three orders of magnitude following shutdowns, start-ups, rapid power changes, and depressurizations. The spike is caused by breaches in the clad which allow water into the gap and allow iodine in the gap to escape. In 1977, the NRC reviewed existing spiking data associated with shutdowns, start-ups, rapid power changes, and depressurizations (Reference 1). The highest spike in this data was 18 μCi/g. Also, the maximum I-131 release from the fuel was 10 Ci per μCi/sec of the equilibrium pre-spike iodine release rate from the fuel.

SRP Model:

The current SRP model for iodine spiking is given in SRP Section 15.1.5, Appendix A, "Radiological Consequences of Main Steam Line Failures Outside

Containment of a PWR," dated July 1981 and SRP Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure (PWR)," dated July 1981. The technical specification limits for coolant activity during plant operation are based on the SRP model. The standard technical specifications have limits of 60 μCi/g and 1 μCi/g. If above 60 μCi/g, the operator must immediately begin to shut down the plant. If above 1 μCi/g, the operator must sample coolant activity and take steps to restore coolant activity to below 1 μCi/g. Extended operation above 1 μCi/g is not permitted.

The SRP states that the dose should be less than the Part 100 dose requirement (300 rem) assuming that the coolant activity is at the upper technical specification limit. In other words, given a MSLB or SGTR accident, the SRP assumes a coolant concentration of 60 μCi/g. The SRP also states that the dose should be less than 10% of the Part 100 dose requirement (30 rem) assuming no pre-existing spike, a coolant concentration at the lower technical specification limit (e.g., 1 μCi/g), and an accident-initiated spiking factor of 500 times the equilibrium release rate.

Post-SRP Work:

As a result of the Ginna SGTR in 1982, RES performed a reassessment of the assumptions used in the SGTR analysis. This work was performed under Generic Issue 67.5.1. In the generic issue resolution memorandum of June 30, 1994 (Reference 2), RES recommended changes to the SRP assumptions for pre-existing spike magnitude, accident-initiated spike magnitude, partition coefficient, pool entrainment, and bypass entrainment. As part of this work, INEL developed and analyzed a database of iodine spikes that are representative of SGTR conditions. Most of the iodine spikes in this database are from reactor trip events. INEL argued that the pressure transient of the reactor trip is similar enough to that of SGTR to use reactor trip spikes in the database (Reference 3).

In 1993, EPRI developed and analyzed a database of iodine spikes that are representative of both SGTR and MSLB conditions (Reference 4). The EPRI database included the SGTR database developed by INEL. EPRI also developed a physically-based model to describe the mechanism of iodine spiking to show the applicability of the SGTR database to MSLB conditions (Reference 5). NRR subsequently contracted with INEL to review EPRI's work. In its subsequent technical evaluation report, INEL questioned the applicability of the SGTR database to MSLB conditions (Reference 6). In addition, INEL identified weaknesses in EPRI's physically-based model.

RES Reassessment:

For the pre-existing spike, the standard technical specification limit is 60 μCi/g. A review of available documentation does not reveal a recorded basis for the use of 60 μCi/g in the standard technical specifications. The highest spike observed has been 18 μCi/g. Based on discussions with current and former NRC staff, it appears that 60 μCi/g was chosen because it is higher than any spike observed, i.e., it is conservative, and because this value could be used in safety analyses and still meet the Part 100 dose requirements.

For the accident-initiated spike, the standard technical specification limit is 1 μCi/g. The SRP states that an initial coolant concentration at the technical specification limit together with a spiking factor of 500 should be used. A review of available documentation does not reveal a recorded basis for the assumption of 1 μCi/g coupled with a spiking factor of 500. This 1981 SRP assumption may have evolved from the 1977 NRC review of spiking data (Reference 1). The 1977 NRC review contains a data table of 69 initial concentrations and spiking factors. In this data table, only 8 of the initial concentrations exceeded 1 μCi/g and only 5 of the spiking factors exceeded 500. INEL and EPRI have compared the spiking factor of 500 with their spiking databases and concluded that a spiking factor of 500 is conservative (References 3 and 4).

At one time, it was considered that the potential for high coolant activity levels during operation should require a dose acceptance criterion less than the Part 100 limit of 300 rem (Reference 7). However, since the current standard technical specifications include a coolant activity limit of 1 μCi/g and because we have accumulated a history of low coolant activity levels during plant operation, there is no longer a compelling basis for a smaller dose acceptance criterion than 300 rem. Note that this is a relaxation of the SRP acceptance criteria, however, it is still in compliance with Part 100. In a SGTR or MSLB analysis, loss of the primary cleanup system is assumed. Assuming loss of the primary cleanup system, the pre-existing spike will have a greater time-integrated concentration than the accident-initiated spike. Therefore, consideration of an accident-initiated spike is unnecessary.

Spiking occurs as a result of rapid power changes and depressurizations. Both MSLB and SGTR accidents result in a scram and a reactor coolant system pressure decrease. Figures 1 and 2 show plots of reactor coolant system pressure versus time for postulated MSLB and SGTR accidents for the Surry plant (Reference 8). For the MSLB, the pressure decreases from 2250 psia to 860 psia in about 2 minutes. For the SGTR, the pressure decreases to 1550 psia in about 6 minutes followed by operator depressurization to 1000 psia in about 30 minutes. The difference in final pressure for the MSLB and SGTR accidents (860 psia and 1000 psia) is not sufficient to affect the spike shape or magnitude. Also, Reference 1 lists spiking data for 69 spikes. This data shows that the time-to-peak varies from 24 minutes to 24 hours with an average time-to-peak of 5 hours. Whether the depressurization takes 2 minutes or 30 minutes will not significantly affect the spike shape or magnitude, because the release from the fuel takes 5 hours. Therefore, the spiking magnitudes will be the same for both the MSLB and SGTR accidents, and the INEL database is representative of both MSLB and SGTR conditions.

<u>Iodine partition coefficient and chemical species</u>:

In a SGTR or MSLB accident, the primary coolant flows through the steam generator on its way to the environment. Some of the primary coolant entering the steam generator flashes (turns to vapor) and the remainder of the water becomes atomized (aerosol), since the primary coolant is initially above the saturation temperature at the steam generator pressure. The SRP assumptions are the following:

dry steam generator: All of the primary leakage is assumed to leave the steam generator.

partially-submerged tubes: The SRP does not give guidance on what to do when the tubes are partially submerged.

completely-submerged tubes: The fraction of iodine transmitted is equal to the fraction of the primary leakage which flashes. Additional credit for scrubbing may be claimed using the models in NUREG-0409 (Reference 9). Any iodine transferred to the steam generator water will become airborne at a rate which is a function of the steaming rate and the iodine partition coefficient. The SRP also states that an iodine partition coefficient of 100 between the steam generator water and vapor phases is conservative.

For a dry steam generator, no credit can be given for scrubbing. However, credit can be given for dilution in the steam generator volume where detailed calculations provide estimates of this dilution. For partially-submerged tubes, it is conservative to assume no scrubbing since the height of the break with respect to the water level is unknown. For completely-submerged tubes, credit can be given for scrubbing and dilution.

Work on Generic Issue 67.5.1 included a reassessment of the assumptions for scrubbing and dilution for completely-submerged tubes of Westinghouse and Combustion Engineering steam generators (Reference 2). This reassessment was based on the following information that was produced since the SRP was issued in July 1981: (1) data from the MB-2 Steam Generator Transient Response Program and (2) laboratory and plant data on iodine partition coefficients under prototypic conditions. This reassessment recommended values for three parameters as follows: a partition coefficient of 35 (mass basis), a pool entrainment of .005%, and a bypass entrainment of .001%. Reference 2 explains how to use these three parameters to conservatively calculate the amount of iodine leaving the steam generator.

The basis of the SRP statement that "a partition coefficient of 100 is conservative" appears to be Figure 5 of NUREG-0409. This figure gives experimentally-measured partition coefficients for elemental iodine. None of the experimentally-measured partition coefficients in this figure were less than 100.

The partition coefficients in NUREG-0409 are given on a volume basis. At a steam generator pressure of 1000 psia, the volume-basis partition coefficient is equal to 20 times the mass-basis partition coefficient. Therefore, the mass-basis partition coefficient of 35 is equal to a volume-basis partition coefficient of 700. The recommended mass-basis coefficient of 35 is a relaxation with respect to the SRP's volume-basis coefficient of 100.

Plateout in the steam generator, steam lines, ADV's, or MSSV's:

The SRP assumes that there is no plateout in the steam generator, steam lines, ADVs, or MSSVs. Based on our experience with the VICTORIA fission-product release and transport code, we believe that an assumption of no plateout is very conservative. Credit for plateout may be given where detailed supporting calculations are provided.

References:

1. "Iodine Spiking in BWR and PWR Coolant Systems," W.F. Pasedag, Paper presented at the ANS Thermal Reactor Safety Meeting, Sun Valley, ID, CONF-770708, 1977

2. Resolution of GI 67.5.1, "Reassessment of SGTR Radiological Consequences," Memorandum from E.S. Beckjord to J.M. Taylor, June 30, 1994

3. "The Iodine Spike Release Rate during a Steam Generator Tube Rupture," J.P. Adams and C.L. Atwood, Nuclear Technology, Vol. 94, June 1991

4. EPRI Draft Report TR-103680, "Review of Iodine Spike Data from PWR Plants in Relation to SGTR with MSLB," December 1993

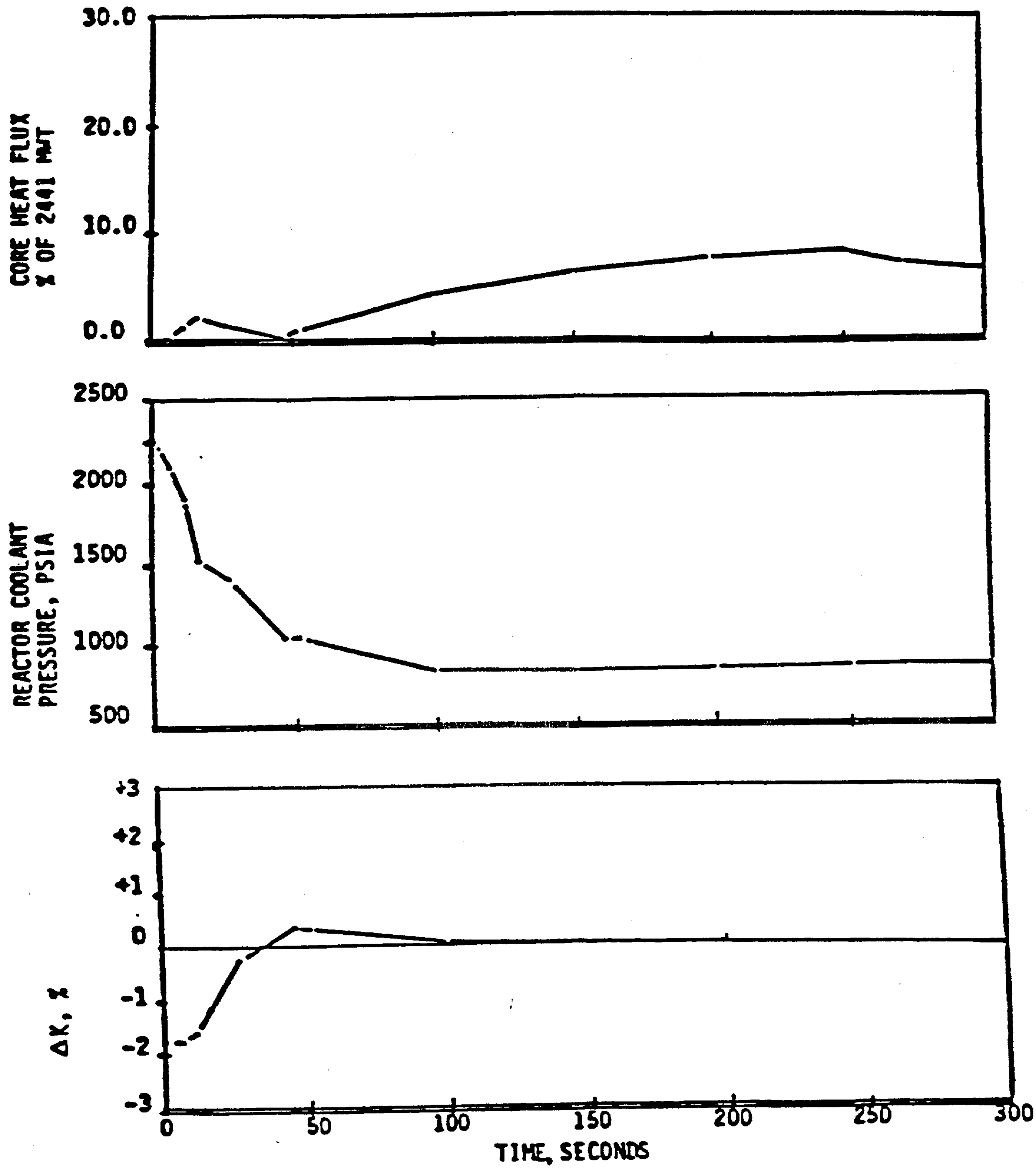
5. EPRI Draft Report, "An Iodine Spiking Model for Pressurized Water Reactor Analysis," October 1995

6. INEL Draft Technical Evaluation Report of Two Iodine Spiking Reports: "Empirical Study of Iodine Spiking in PWR Power Plants," TR-103680, Rev. 1 and "An Iodine Spiking Model For Pressurized Water Reactor Analysis, Volume 1: Theory Manual," January 1996

7. "Radiological Consequences of a Main Steam Line Failure with Large Steam Generator Tube Leaks," H.M. Fontecilla and B.K. Grimes, Paper presented at the ANS Thermal Reactor Safety Meeting, Sun Valley, ID, CONF-770708, 1977

8. "Surry Power Station Units 1 & 2, Updated Final Safety Analysis Report," Volume V, Revision 19, February 1993

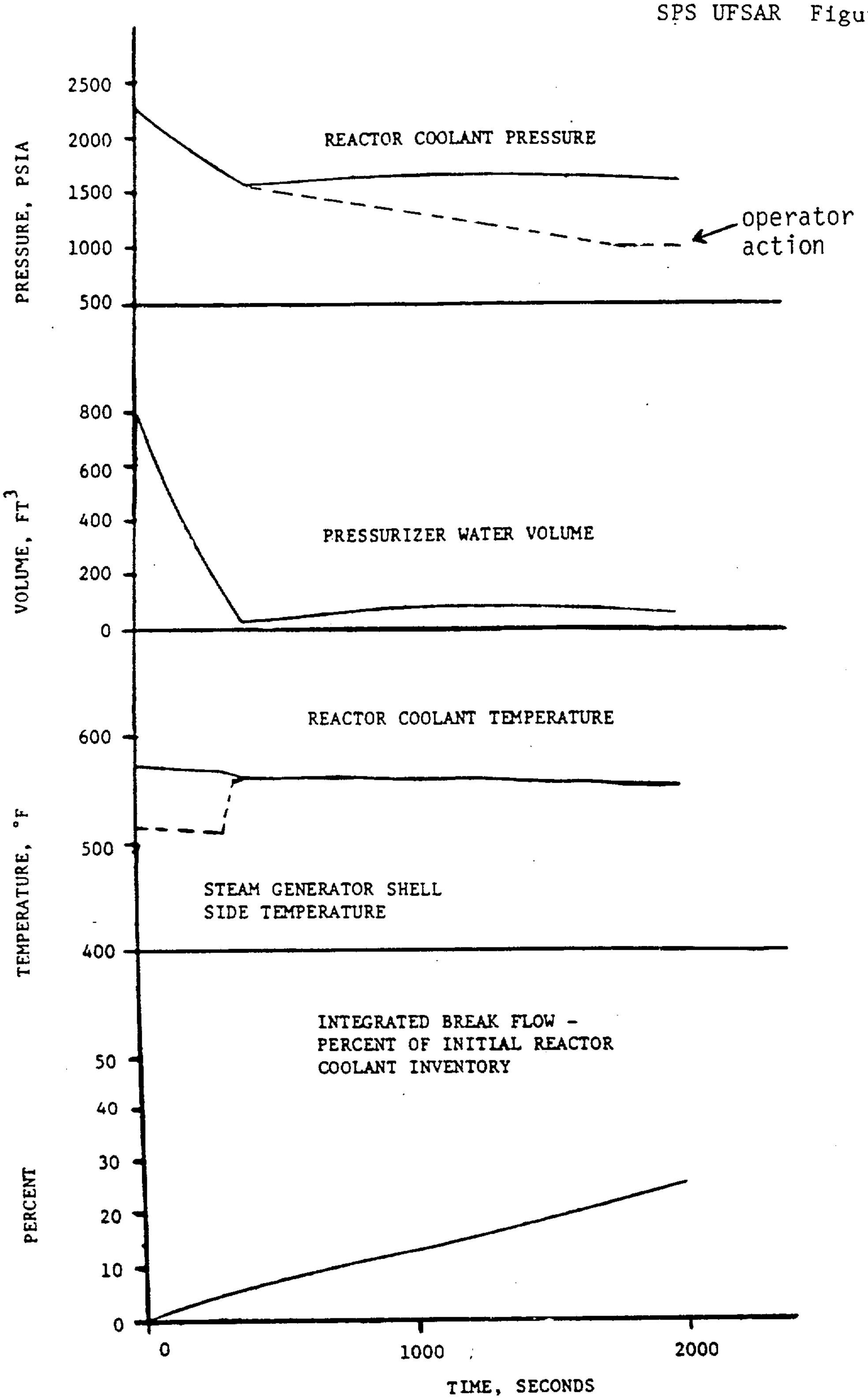
9. NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," January 1978



STEAM LINE BREAK AT EXIT OF STEAM GENERATOR WITHOUT OUTSIDE POWER

(CASE B)

Figure 1



TUBE RUPTURE STUDY
CASE 4

Figure 2

ATTACHMENT B

Monte Carlo Method for Analyzing the Radiological Consequences for a Main Steam Line Break

In an August 18, 1995, memorandum from RES (W. Hodges) to NRR (A. Thadani), it was agreed that RES would pursue development of alternative methods for radiological consequence assessment for a Main Steam Line Break (MSLB). The following describes an alternative method, its use, and underlying assumptions.

Background:

Standard Review Plan Section 15.1.5, Appendix A, "Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR," dated July 1981 describes the current methodology for analyzing the radiological consequences for a MSLB. The current methodology is based on conservative assumptions and requires calculating a dose at three locations, namely, the exclusion area boundary, low population zone distance, and control room.

The NRC is considering revising requirements for plugging and sleeving degraded tubes. If the requirements are relaxed, then there may be a greater probability of additional leakage through the tubes for the MSLB. This additional leakage may cause doses calculated using the current methodology to exceed the acceptance criteria.

EPRI and Dr. D. Powers (ACRS) have proposed an alternative approach for analyzing the consequences of a MSLB (References 1 and 2). This alternative approach involves replacing bounding assumptions for each of the individual elements of the current methodology with an integrated probabilistic treatment. Through a Monte Carlo analysis, a probability distribution of doses can be constructed. This probability distribution of dose can then be used as the basis for licensing in lieu of the bounding deterministic calculation.

Current Methodology:

The current methodology utilizes the following basic equation for calculating an inhalation dose:

$$\text{Dose} = \text{coolant concentration} \times \text{leak rate} \times \text{dispersion factor } (X/Q) \times \text{breathing rate} \times \text{time} \times \text{dose conversion factor}$$

In selecting each of the terms in this equation, conservative assumptions are made. The following table summarizes the current methodology and its assumptions.

Analysis Step	Assumption
primary coolant activity concentration Case 1: pre-existing spike Case 2: accident-initiated spike	 60 μCi/g 1 μCi/g, followed by release rate of 500x equilibrium release rate
primary coolant cleanup system	system is unavailable
primary-to-secondary leak rate	1 gpm
activity transport to environment	all activity that leaks from the primary becomes airborne
X/Q	Regulatory Guide 1.145
breathing rate	3.47×10^{-4} m^3/sec
dose conversion factor	1.48×10^{6} rem/Ci
acceptance criteria Case 1: pre-existing spike Case 2: accident-initiated spike	 300 rem thyroid/25 rem whole body 30 rem thyroid/2.5 rem whole body

<u>Alternative Monte Carlo Methodology</u>:

An alternative method for determining the radiological consequences for MSLB has been developed. This method is a Monte Carlo approach and is implemented in the code SG.BAS (attached) written in Microsoft QUICKBASIC Version 4.5. SG.BAS calculates a dose distribution for an individual at the exclusion area boundary. A dose distribution calculated by SG.BAS is given in Figure 1. This dose distribution was calculated using the EPRI-proposed spiking magnitude distribution for PWRs (Reference 3), the dispersion factor distribution for Site A of WASH-1400 (Reference 4), and a sample size of 40,000 histories. From Figure 1, it can be seen that given a 100 gpm leak, there is a 2% probability of exceeding Part 100. Hence, this method provides a direct quantification of the uncertainty in exceeding Part 100. Figure 2 was produced by performing a calculation for each of five leak rates. Using a possible acceptance criterion of less than 1% probability of exceeding Part 100 (99th percentile), a maximum leak rate of 72 gpm is acceptable.

For comparison, we calculated the maximum acceptable leak rate using the SRP methodology and the standard technical specification pre-existing spike concentration of 60 μCi/g. The result is a maximum leak rate of 20 gpm.

The proposed integrated probabilistic methodology solves the same basic equation as the current methodology, but for each of the inputs to the equation the probabilistic method replaces the existing single conservative value with a probability distribution of values. For example, the current methodology assumes a 60 μCi/g coolant concentration and a 95th percentile dispersion factor. In SG.BAS, these conservative assumptions are replaced with a probability distribution of coolant concentrations and a probability distribution of dispersion factors. By replacing conservative assumptions with probability distributions, the result of the dose calculation is a probability distribution of dose. This probability distribution of dose is thus a quantification of the uncertainty in the dose analysis. Figures 3 and 4 show the steps and flow chart for the proposed Monte Carlo methodology.

Implementation of Proposed Monte Carlo Methodology

The following is a delineation of the issues related to implementation of the proposed Monte Carlo methodology and the recommended resolution:

Use of the Monte Carlo methodology to treat the two spiking cases (i.e., the pre-existing spike and the accident-initiated spike):

As discussed in Attachment A, two spiking cases are considered in the Standard Review Plan, namely, the pre-existing spike and the accident-initiated spike. The dose limit for the pre-existing spike is 300 rem. The dose limit for the accident-initiated spike is 30 rem. RES recommended in Attachment A that the 300 rem limit be applied to the accident-initiated spike. Since the pre-existing spike has a greater time-integrated concentration than the accident-initiated spike, the pre-existing spike would be limiting. Therefore, it is not necessary to evaluate the consequences of the accident-initiated spike case. For the pre-existing spike case, a probability distribution of spike magnitudes is used.

Construction of a probability distribution for the spike magnitude:

Two options for constructing a distribution are available. The first option is to combine the distributions for gap activity and number of defected fuel rods to get a probability distribution for spike magnitude (Reference 2). The second option is to use spiking data taken at commercial nuclear power plants (Reference 3). It should be noted that use of the Reference 3 spiking data is limiting in that the largest spikes in this data occurred before 1975 and more recent data reflecting newer fuel design and manufacturing indicates lower spikes. Additionally, the ongoing accumulation of plant data industry-wide may be evaluated for consistency with assumed iodine spiking distributions.

Technical specifications requirements related to iodine spiking:

The substitution of an iodine spiking distribution for the previous single upper limit value of 60 μCi/g should be considered in the evaluation of appropriate plant surveillance measures.

Construction of a probability distribution for primary-to-secondary leak rate:

The leak rate is a function of the defects in the tubes and the pressure difference between the primary and the secondary system. From discussions with NRR staff, it appears that licensees will be predicting the defects in the tubes at end of cycle. For simplicity, the leak rate used in the Monte Carlo analysis should assume that the plant is at the end of cycle. The pressure difference across the tubes will decrease as the primary system pressure decreases from normal operating pressure, since the operators will depressurize the primary system following the MSLB. A simplifying assumption would be that the pressure difference is at a maximum for the entire period of the analysis. Therefore, the probability distribution for leak rate should be for the defects predicted at end of cycle and with maximum pressure difference across the tubes.

Construction of a probability distribution for the dispersion factor:

Hourly site meteorology data are available for each plant. For each hour of data, a centerline dispersion factor can be calculated. A probability distribution can be constructed using these hourly dispersion factors. It is recommended that the hourly dispersion factors for a period of one year (8760 hourly dispersion factors), or integral multiples thereof, be used to construct the probability distribution.

As noted by NRR (J. Hayes) in a meeting with RES (J. Schaperow) on May 28, 1996, a refinement of the RES treatment of dispersion factors would utilize a chronological listing of hourly dispersion factors covering a period of one year. For the two-hour exclusion area boundary dose calculation, one would randomly select a dispersion factor from the chronological listing for the first hour of the dose calculation. The subsequent dispersion factor in the chronological listing would be used for the second hour.

Treatment of wind direction:

The Part 100 dose is to an individual located at any point on the exclusion area boundary (EAB) for two hours or at the low population zone (LPZ) distance for the duration of the accident. A conservative and easy-to-implement assumption is that the wind always blows toward the individual. This is a reasonable assumption for the EAB dose which is a two-hour dose. However, the LPZ dose is for the duration of the accident. For a MSLB, the duration of the accident is the time it takes to depressurize the primary coolant system. This time could be up to several hours. Over several hours the wind is more likely to change direction. However, the assumption of wind always blowing toward the individual may also be acceptable for the LPZ dose. One possible reason why this may be acceptable is that the analyst may find that the steam generator leak rate will not be limited by the LPZ dose, but will instead be limited by either the EAB dose or the make-up rate of the charging pumps. However, if one did want to take the wind direction into account, a relatively straightforward method is available using sampling of the meteorological data and calculating doses for each of the sectors.

Use of the resulting Monte Carlo dose distribution for regulatory decisions:

Given a MSLB, the Monte Carlo methodology results in a probability distribution of dose. One possible acceptance criterion is that the 99th percentile dose be less than 300 rem. This is similar to what is done with respect to ECCS performance for a LOCA. Given a LOCA, Regulatory Guide 1.157 states that ECCS performance is considered acceptable when the 95th percentile maximum clad temperature is less than 2200 F.

References:

1. Technical Basis for Considering Uncertainties in I-131 Release and Dose Limits for a Postulated Accident, EPRI Draft Report TR-103878, March 1994

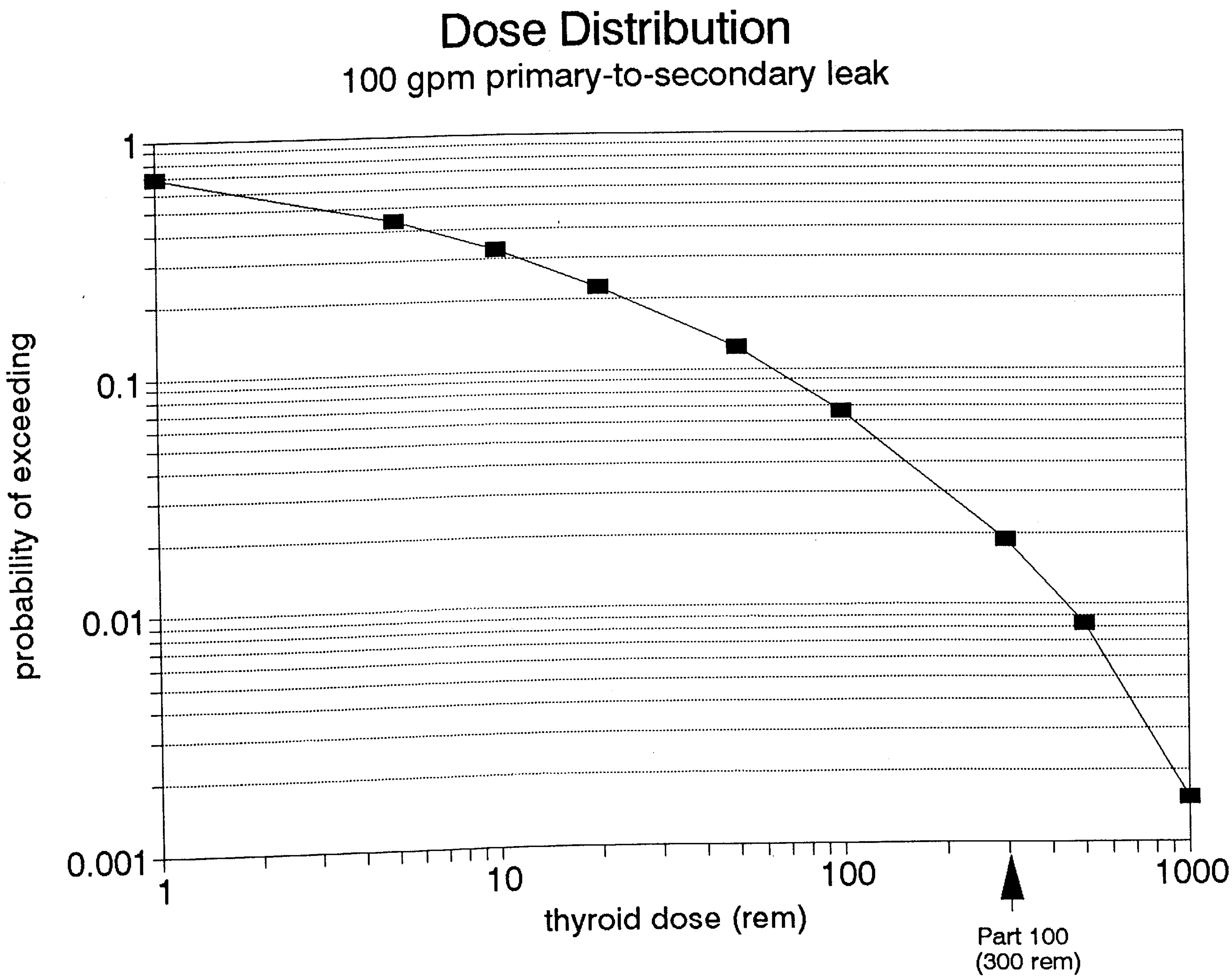
2. Memorandum to W.T. Shack (ACRS) from D.A. Powers (ACRS) dated August 17, 1994

3. "Review of Iodine Spike Data from PWR Power Plants in Relation to SGTR with MSLB," EPRI Draft Report TR-103680, December 1993

4. WASH-1400, "Reactor Safety Study," October 1975

5. "Radiological Consequences of a Main Steam Line Failure with Large Steam Generator Tube Leaks," H.M. Fontecilla and B.K. Grimes, Paper presented at the ANS Thermal Reactor Safety Meeting, Sun Valley, ID, CONF-770708, 1977

Attachment: Monte Carlo Program to
Calculate DBA Dose Distribution



Dose Distribution
100 gpm primary-to-secondary leak
probability of exceeding
1
0.1
0.01
0.001
1
10
100
1000
thyroid dose (rem)
Part 100
(300 rem)

Figure 1

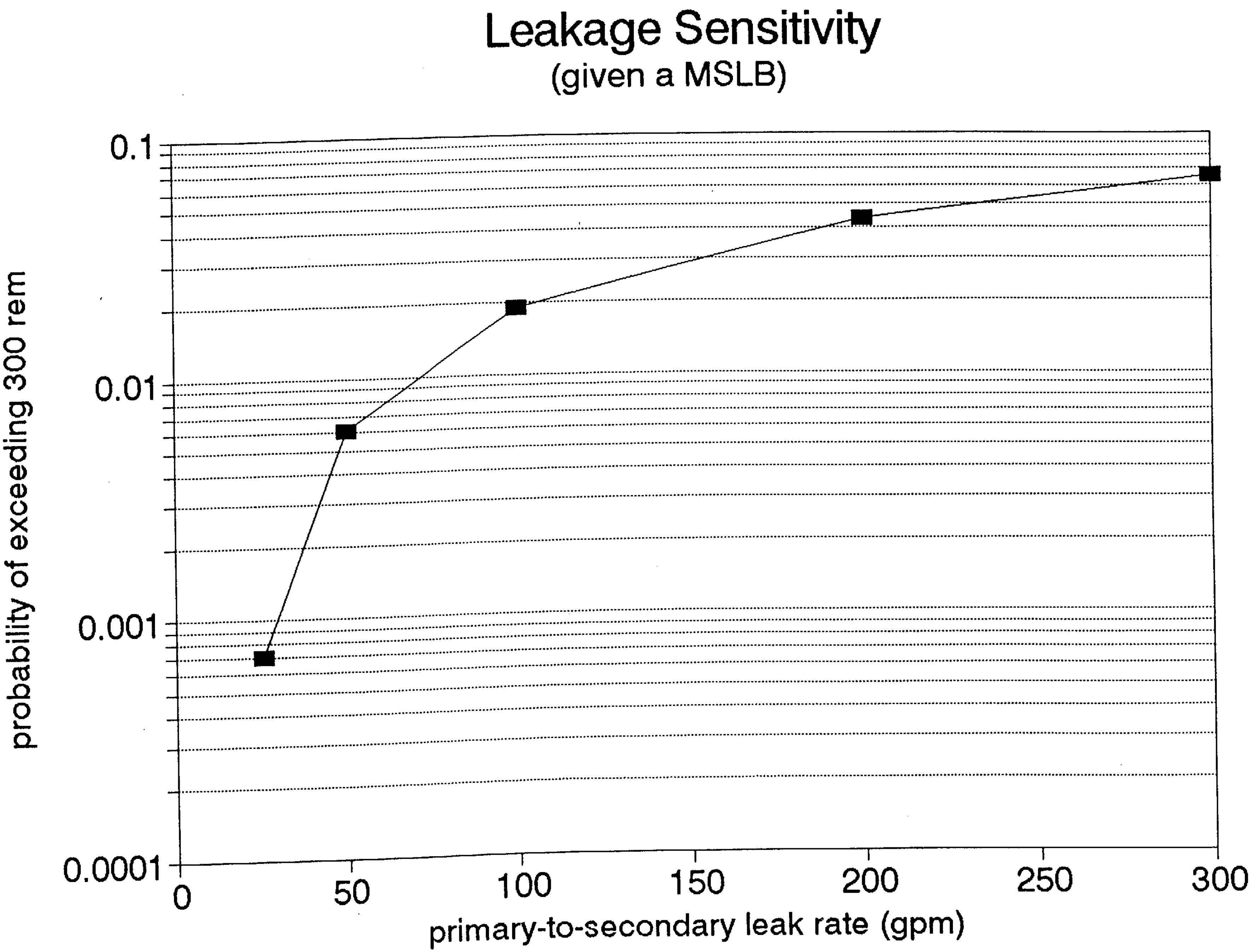

Leakage Sensitivity
(given a MSLB)
probability of exceeding 300 rem
0.1
0.01
0.001
0.0001
0
50
100
150
200
250
300
primary-to-secondary leak rate (gpm)

Figure 2

Steps in Monte Carlo Methodology

Step 1: Select a distribution for each of the following variables: primary coolant activity concentration, primary-to-secondary leak rate, and dispersion factor. (It is not necessary to select distributions for breathing rate or dose conversion factor, since these variables are much less uncertain.)

Step 2: Pick a random number between 0 and 1. Use this random number to select a primary coolant activity concentration from its cumulative distribution function (CDF).

Step 3: Pick a random number between 0 and 1. Use this random number to select a primary-to-secondary leak rate from its CDF.

Step 4: Pick a random number between 0 and 1. Use this random number to select a dispersion factor from its CDF.

Step 5: Calculate a single dose using the concentration, leak rate, and dispersion factor from steps 2, 3, and 4.

Step 6: Repeat steps 2 through 5 to obtain the desired number of histories.

Step 7: Use the results of step 6 to construct a CDF of doses.

Step 8: Compare the CDF of doses to the acceptance criteria.

Figure 3

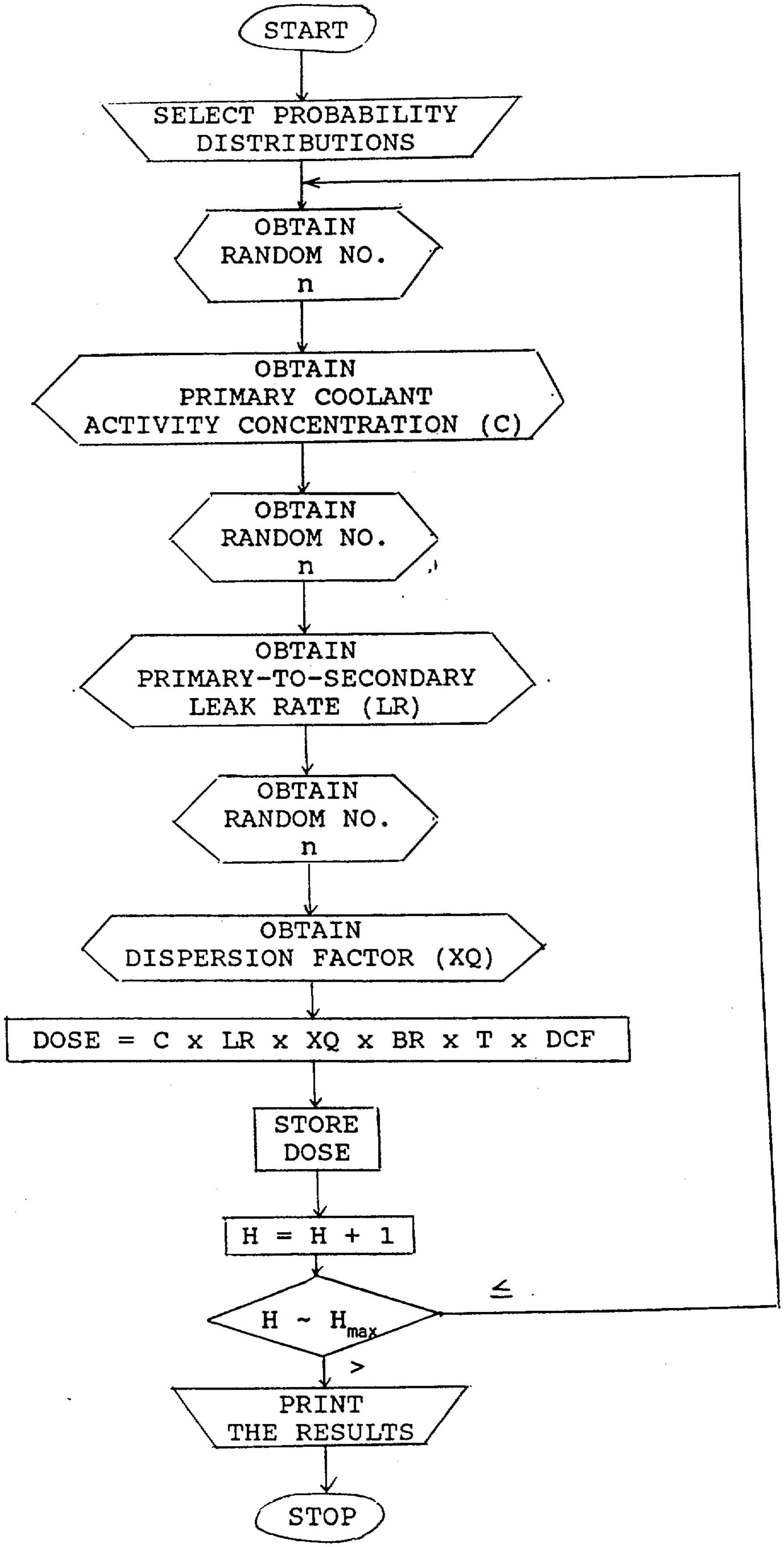

START
SELECT PROBABILITY DISTRIBUTIONS
OBTAIN RANDOM NO. n
OBTAIN PRIMARY COOLANT ACTIVITY CONCENTRATION (C)
OBTAIN RANDOM NO. n
OBTAIN PRIMARY-TO-SECONDARY LEAK RATE (LR)
OBTAIN RANDOM NO. n
OBTAIN DISPERSION FACTOR (XQ)
DOSE = C x LR x XQ x BR x T x DCF
STORE DOSE
H = H + 1
H ~ H max
≤
>
PRINT THE RESULTS
STOP

Figure 4

/

```
'
'
'     *************************************************************
'     *   SG.BAS     5/28/96                                      *
'     *   Monte carlo program to calculate DBA dose distribution.  *
'     *************************************************************
'
'     This code was revised 5/28/96 to change the sampling routine for
'     peak iodine concentration from "I=191*A" to "I=191*A+1" to make
'     the probability of selecting the highest spike 1/191, not 0.
'
'
'
DIM PROB(191), PEAK(191), PEAK1(191), CONC(191), RELRATE(191)
DIM CDFXQ(48), XQ(48)
DIM DOSE(1000)
'
K1 = 0: K2 = 0: K3 = 0: K4 = 0: K5 = 0: K6 = 0: K7 = 0: K8 = 0: K9 = 0: K10 = 0
CLS
'
'  ****************************************************
'  *  CDF for peak iodine concentration WITH clean-up  *
'  ****************************************************
'
'  PROB(I) = cumulative probability
'  PEAK(I) = peak iodine concentration with clean-up (uCi/g)
'
FOR I = 1 TO 191
PROB(I) = I / 191
NEXT I
  PEAK(1) = .000215
  PEAK(2) = .00003
  PEAK(3) = .000495
  PEAK(4) = .00068
  PEAK(5) = .00074
  PEAK(6) = .000788
  PEAK(7) = .0008
  PEAK(8) = .000934
  PEAK(9) = .00189
  PEAK(10) = .002
  PEAK(11) = .003
  PEAK(12) = .00388
  PEAK(13) = .00525
  PEAK(14) = .00607
  PEAK(15) = .00893
  PEAK(16) = .0101
  PEAK(17) = .0108
  PEAK(18) = .0264
  PEAK(19) = .0306
  PEAK(20) = .0315
  PEAK(21) = .032
  PEAK(22) = .033
  PEAK(23) = .0342
  PEAK(24) = .0355
  PEAK(25) = .036
  PEAK(26) = .0393
  PEAK(27) = .041
  PEAK(28) = .0514
```

```
PEAK(29) = .0564
PEAK(30) = .0567
PEAK(31) = .0572
PEAK(32) = .0574
PEAK(33) = .059
PEAK(34) = .064
PEAK(35) = .0666
PEAK(36) = .067
PEAK(37) = .0698
PEAK(38) = .073
PEAK(39) = .0747
PEAK(40) = .0751
PEAK(41) = .0877
PEAK(42) = .0896
PEAK(43) = .091
PEAK(44) = .1
PEAK(45) = .11
PEAK(46) = .111
PEAK(47) = .116
PEAK(48) = .133
PEAK(49) = .135
PEAK(50) = .135
PEAK(51) = .136
PEAK(52) = .14
PEAK(53) = .145
PEAK(54) = .149
PEAK(55) = .15
PEAK(56) = .153
PEAK(57) = .155
PEAK(58) = .159
PEAK(59) = .16
PEAK(60) = .165
PEAK(61) = .17
PEAK(62) = .17
PEAK(63) = .171
PEAK(64) = .179
PEAK(65) = .179
PEAK(66) = .18
PEAK(67) = .182
PEAK(68) = .19
PEAK(69) = .192
PEAK(70) = .194
PEAK(71) = .198
PEAK(72) = .222
PEAK(73) = .23
PEAK(74) = .23
PEAK(75) = .236
PEAK(76) = .236
PEAK(77) = .237
PEAK(78) = .242
PEAK(79) = .246
PEAK(80) = .25
PEAK(81) = .25
PEAK(82) = .258
PEAK(83) = .259
PEAK(84) = .26
PEAK(85) = .265
PEAK(86) = .275
```

```
PEAK(87) = .28
PEAK(88) = .299
PEAK(89) = .3
PEAK(90) = .3
PEAK(91) = .302
PEAK(92) = .304
PEAK(93) = .314
PEAK(94) = .325
PEAK(95) = .331
PEAK(96) = .35
PEAK(97) = .388
PEAK(98) = .395
PEAK(99) = .4
PEAK(100) = .403
PEAK(101) = .41
PEAK(102) = .425
PEAK(103) = .43
PEAK(104) = .439
PEAK(105) = .44
PEAK(106) = .448
PEAK(107) = .47
PEAK(108) = .481
PEAK(109) = .484
PEAK(110) = .52
PEAK(111) = .526
PEAK(112) = .531
PEAK(113) = .535
PEAK(114) = .535
PEAK(115) = .545
PEAK(116) = .56
PEAK(117) = .58
PEAK(118) = .6
PEAK(119) = .6
PEAK(120) = .62
PEAK(121) = .631
PEAK(122) = .65
PEAK(123) = .679
PEAK(124) = .693
PEAK(125) = .704
PEAK(126) = .728
PEAK(127) = .764
PEAK(128) = .8
PEAK(129) = .812
PEAK(130) = .824
PEAK(131) = .845
PEAK(132) = .858
PEAK(133) = .874
PEAK(134) = .9
PEAK(135) = .918
PEAK(136) = .922
PEAK(137) = .93
PEAK(138) = .937
PEAK(139) = 1.02
PEAK(140) = 1.05
PEAK(141) = 1.05
PEAK(142) = 1.11
PEAK(143) = 1.11
PEAK(144) = 1.15
```

```
PEAK(145) = 1.18
PEAK(146) = 1.2
PEAK(147) = 1.22
PEAK(148) = 1.44
PEAK(149) = 1.44
PEAK(150) = 1.6
PEAK(151) = 1.66
PEAK(152) = 1.7
PEAK(153) = 1.72
PEAK(154) = 1.8
PEAK(155) = 1.8
PEAK(156) = 1.98
PEAK(157) = 1.99
PEAK(158) = 1.99
PEAK(159) = 2.04
PEAK(160) = 2.16
PEAK(161) = 2.35
PEAK(162) = 2.5
PEAK(163) = 2.61
PEAK(164) = 2.65
PEAK(165) = 2.8
PEAK(166) = 3!
PEAK(167) = 3!
PEAK(168) = 3!
PEAK(169) = 3.12
PEAK(170) = 3.32
PEAK(171) = 5.07
PEAK(172) = 5.14
PEAK(173) = 5.18
PEAK(174) = 5.2
PEAK(175) = 5.5
PEAK(176) = 5.57
PEAK(177) = 6!
PEAK(178) = 6!
PEAK(179) = 6.8
PEAK(180) = 7.4
PEAK(181) = 7.43
PEAK(182) = 8.2
PEAK(183) = 8.3
PEAK(184) = 8.97
PEAK(185) = 10.2
PEAK(186) = 11!
PEAK(187) = 12!
PEAK(188) = 14.4
PEAK(189) = 15.5
PEAK(190) = 18.1
PEAK(191) = 18.3
'
'
'  ********************************************************
'  *  CDF for peak iodine concentration WITHOUT clean-up  *
'  ********************************************************
'
'  LAMBDA    = removal constant for clean-up system (/hr)
'  T         = time of the peak (hr)
'  PEAK1(I)   = peak iodine concentration without clean-up (uCi/g)
'
'  To convert from uCi/g to uCi/gal, multiply by
```

```
'  455 g/lbm, 62.5lbm/ft3(@STP), .13368 ft3/gal  ==> 3801.5 g/gal
'
'  CONC(I)     = peak iodine concentration without clean-up (uCi/gal)
'
'
'
'
'  Since this code does not currently include a probability distribution
'  of leak rates, this block of code asks the user to input a single
'  leak rate.  This block of code also uses this leak rate to produce a
'  probabilistic distribution of release rates.
'
'  LEAKRATE = primary-to-secondary leak rate (gal/min)
'  RELRATE(I) = iodine release rate (Ci/min)
'
LAMBDA = .1
T = 6!
INPUT "Leak rate in gpm:   ", LRINP
LEAKRATE = LRINP
'
'
'OPEN "TEST1.DAT" FOR OUTPUT AS #1
OPEN "TEST2.DAT" FOR OUTPUT AS #2
'OPEN "TEST3.DAT" FOR OUTPUT AS #3
'PRINT #1, "PROB(I)  PEAK(I)  PEAK1(I)  CONC(I)  RELRATE(I)"
'PRINT #1, "           uCi/g    uCi/g    uCi/gal    Ci/min"
'
'
FOR I = 1 TO 191
PEAK1(I) = PEAK(I) * EXP(LAMBDA * T)
CONC(I) = PEAK1(I) * 3801.5
RELRATE(I) = CONC(I) * LEAKRATE
'
'PRINT #1, USING "##.### ##.###### ##.###### ##.##^^^^ ##.##^^^^"; PROB(I); PEAK
'
NEXT I
'
'
'  ********************************************
'  *  CDF for primary-to-secondary leak rate  *
'  ********************************************
'
'  This code does not currently include a probability distribution
'  of leak rates.
'
'
'
'  ****************************************
'  *  CDF for X/Q for Site A of WASH-1400  *
'  ****************************************
'
CDFXQ(1) = .0155:        XQ(1) = 5.55E-06
CDFXQ(2) = .0269:        XQ(2) = .0000064
CDFXQ(3) = .0412:        XQ(3) = 7.57E-06
CDFXQ(4) = .0629:        XQ(4) = 9.25E-06
CDFXQ(5) = .0866:        XQ(5) = .0000119
CDFXQ(6) = .0901:        XQ(6) = .0000149
CDFXQ(7) = .1173:        XQ(7) = .0000167
CDFXQ(8) = .1179:        XQ(8) = .0000172
```

```
CDFXQ(9) = .1194:        XQ(9) = .0000203
CDFXQ(10) = .1219:       XQ(10) = .0000248
CDFXQ(11) = .1518:       XQ(11) = .0000278
CDFXQ(12) = .1567:       XQ(12) = .0000289
CDFXQ(13) = .1588:       XQ(13) = .0000319
CDFXQ(14) = .1611:       XQ(14) = .0000333
CDFXQ(15) = .1638:       XQ(15) = .0000394
CDFXQ(16) = .1664:       XQ(16) = .0000446
CDFXQ(17) = .17:         XQ(17) = .0000481
CDFXQ(18) = .1731:       XQ(18) = .0000619
CDFXQ(19) = .1765:       XQ(19) = .0000743
CDFXQ(20) = .2045:       XQ(20) = .0000829
CDFXQ(21) = .2237:       XQ(21) = .0000833
CDFXQ(22) = .2279:       XQ(22) = .0000867
CDFXQ(23) = .2411:       XQ(23) = .0000957
CDFXQ(24) = .2654:       XQ(24) = .000113
CDFXQ(25) = .2941:       XQ(25) = .000138
CDFXQ(26) = .2995:       XQ(26) = .000144
CDFXQ(27) = .3191:       XQ(27) = .000147
CDFXQ(28) = .3285:       XQ(28) = .00017
CDFXQ(29) = .3819:       XQ(29) = .000177
CDFXQ(30) = .4192:       XQ(30) = .000178
CDFXQ(31) = .4316:       XQ(31) = .000201
CDFXQ(32) = .4366:       XQ(32) = .000223
CDFXQ(33) = .4627:       XQ(33) = .000246
CDFXQ(34) = .5152:       XQ(34) = .000249
CDFXQ(35) = .577:        XQ(35) = .000295
CDFXQ(36) = .6145:       XQ(36) = .000316
CDFXQ(37) = .6162:       XQ(37) = .000347
CDFXQ(38) = .6178:       XQ(38) = .0004
CDFXQ(39) = .6681:       XQ(39) = .000415
CDFXQ(40) = .6941:       XQ(40) = .000416
CDFXQ(41) = .6993:       XQ(41) = .000433
CDFXQ(42) = .7025:       XQ(42) = .000473
CDFXQ(43) = .7082:       XQ(43) = .000578
CDFXQ(44) = .7578:       XQ(44) = .000694
CDFXQ(45) = .7707:       XQ(45) = .000743
CDFXQ(46) = .8521:       XQ(46) = .000885
CDFXQ(47) = .9164:       XQ(47) = .00124
CDFXQ(48) = 1!:          XQ(48) = .00208
'
'
'   ******************************
'   *  Do monte carlo sampling   *
'   ******************************
'
INPUT "Number of histories:  ", M
'
RANDOMIZE TIMER
'
FOR L = 1 TO M
'   ****************************************************
'   *  Sample from the primary coolant concentrations  *
'   ****************************************************
'
'   Note:  For increased speed, this code instead samples from the
'   release rates, since the primary-to-secondary leak rate is a
'   single value.
```

```
'
A = RND
'PRINT A
I = 191 * A + 1
J = FIX(I)
RR = RELRATE(J)
'PRINT "A, I, J, RR"
'PRINT A, I, J, RR
'
'  ******************************************************
'  *  Sample from for primary-to-secondary leak rates   *
'  ******************************************************
'
'  This code does not currently include a probability distribution
'  of leak rates.
'
'   **************************
'   *  Sample from the XQ's  *
'   **************************
'
A = RND
FOR I = 1 TO 47
IF (A < CDFXQ(I)) THEN GOTO 10
NEXT I
'
10 XQ = XQ(I - 1) + ((A - CDFXQ(I - 1)) / (CDFXQ(I) - CDFXQ(I - 1))) * (XQ(I) -
'PRINT "A, I - 1, CDFXQ(I - 1), XQ"
'PRINT A, I - 1, CDFXQ(I - 1), XQ
'
'  ************************
'  *  Calculate the dose  *
'  ************************
'
'
'
BR = .000347
TIME = 120!
DCF = 1480000!
'
DOSE = RR * .000001 * XQ * BR * TIME * DCF
'
'PRINT #1, USING "##.##^^^^ uCi/min   ##.##^^^^ rem"; RR; DOSE
'
'  ***************************
'  *  Put the dose in a bin  *
'  ***************************
'
IF DOSE < 1! THEN K1 = K1 + 1: GOTO 30
IF DOSE < 5! THEN K2 = K2 + 1: GOTO 30
IF DOSE < 10! THEN K3 = K3 + 1: GOTO 30
IF DOSE < 20! THEN K4 = K4 + 1: GOTO 30
IF DOSE < 50! THEN K5 = K5 + 1: GOTO 30
IF DOSE < 100! THEN K6 = K6 + 1: GOTO 30
IF DOSE < 300! THEN K7 = K7 + 1: GOTO 30
IF DOSE < 500! THEN K8 = K8 + 1: GOTO 30
IF DOSE < 1000! THEN K9 = K9 + 1: GOTO 30
K10 = K10 + 1
'
```

```
30 'PRINT "DOSE"
'
'PRINT DOSE
'
'PRINT "K1, K2, K3, K4, K5, K6, K7, K8, K9, K10"
'PRINT K1, K2, K3, K4, K5, K6, K7, K8, K9, K10
'
'PRINT #3, L
PRINT L
NEXT L
'PRINT "put output in test2.dat"
PRINT #2, USING "    < 1 rem    #####"; K1
PRINT #2, USING "    1-5 rem    #####"; K2
PRINT #2, USING "   5-10 rem    #####"; K3
PRINT #2, USING "  10-20 rem    #####"; K4
PRINT #2, USING "  20-50 rem    #####"; K5
PRINT #2, USING " 50-100 rem    #####"; K6
PRINT #2, USING "100-300 rem    #####"; K7
PRINT #2, USING "300-500 rem    #####"; K8
PRINT #2, USING "500-1000 rem   #####"; K9
PRINT #2, USING "  > 1000 rem   #####"; K10
'
'
PRINT #2, " "
PRINT #2, " "
'
PRINT #2, USING "  >   1 rem    #####"; K2 + K3 + K4 + K5 + K6 + K7 + K8 + K9 + K
PRINT #2, USING "  >   5 rem    #####"; K3 + K4 + K5 + K6 + K7 + K8 + K9 + K10
PRINT #2, USING "  >  10 rem    #####"; K4 + K5 + K6 + K7 + K8 + K9 + K10
PRINT #2, USING "  >  20 rem    #####"; K5 + K6 + K7 + K8 + K9 + K10
PRINT #2, USING "  >  50 rem    #####"; K6 + K7 + K8 + K9 + K10
PRINT #2, USING "  > 100 rem    #####"; K7 + K8 + K9 + K10
PRINT #2, USING "  > 300 rem    #####"; K8 + K9 + K10
PRINT #2, USING "  > 500 rem    #####"; K9 + K10
PRINT #2, USING "  > 1000 rem   #####"; K10
'
STOP
END
```