NRC FORM 464 Part I
(6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) / PRIVACY ACT (PA) REQUEST

FOIA/PA: 2000-0255

RESPONSE NUMBER: 1

RESPONSE TYPE: ✓ FINAL ☐ PARTIAL

REQUESTER: Roger Gorman

DATE: JUN 20 2000

PART I. -- INFORMATION RELEASED

☐ No additional agency records subject to the request have been located.

☐ Requested records are available through another public distribution program. See Comments section.

☐ APPENDICES: Agency records subject to the request that are identified in the listed appendices are already available for public inspection and copying at the NRC Public Document Room.

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☐ Certain information in the requested records is being withheld from disclosure pursuant to the exemptions described in and for the reasons stated in Part II.

☐ This determination may be appealed within 30 days by writing to the FOIA/PA Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Clearly state on the envelope and in the letter that it is a "FOIA/PA Appeal."

PART I.C COMMENTS (Use attached Comments continuation page if required)

SIGNATURE - FREEDOM OF INFORMATION ACT AND PRIVACY ACT OFFICER

For Carol Ann Reed

APPENDIX A

RECORDS BEING RELEASED IN THEIR ENTIRETY
(If copyrighted identify with*)

NUMBER	DATE	DESCRIPTION/PAGES
1.	6/30/94	Memo for J. Taylor from E. Beckjord, subject: Resolution of GI 67.5.1, "Reassessment of SGTR Radiological Consequences," (27 pgs.).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

JUN 30 1994

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

SUBJECT: RESOLUTION OF GI 67.5.1 "REASSESSMENT OF SGTR RADIOLOGICAL CONSEQUENCES"

Summary

The subject generic safety issue was identified to address a concern raised following the January 1982 SGTR event at the R. E. Ginna nuclear power plant. Thi concern questioned the validity of Standard Review Plan (SRP) assumptions in Section 15.6.3 regarding radiological release dose calculations performed to predict the consequences of the design basis SGTR. A study was initiated under GI 67.5.1 and the research discussed below was conducted. RES is now proposing to close out this GI after having accomplished the objective of identifying improvements to the calculational methods in SRP Section 15.6.3. The ACRS was briefed on this issue; however, we did not request nor did they provide a letter on the issue.

The enclosed report, "Resolution of GI 67.5.1 'Reassessment of SGTR Radiological Consequences'", provides the results of our effort to evaluate the methodology for calculating the radiological releases from an SGTR event. Justification is provided to remove some ambiguities and decrease somewhat conservatism in the SRP method; furthermore important limitations of the data are noted. The enclosed report discusses proposed changes to the SRP; however, none will be made at this time. Instead, NRR will consider these recommendations in conjunction with ongoing proposed rulemaking activities related to establishing degradation-specific steam generator sureveillance and maintenance requirements. NRR will also consider the technical findings in the reassessment report as appropriate in its generic and plant specific safety evaluation pending completion of the rulemaking.

Discussion

Three major parameters must be specified to calculate radiological dose releases from an SGTR accident, viz. (i) iodine partitioning, (ii) liquid carryover, and (iii) iodine spiking. Elaboration is provided on these terms in the enclosed report. The SRP includes guidance on each of these parameters, but provides insufficient technical justification. To improve the technical basis for staff review of licensees' submittals, three tasks were undertaken as follows:

(a) A joint NRC industry steam generator test program was initiated called the MB-2 Program.

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(b) An iodine chemistry study was performed to obtain data on iodine partitioning under prototypic conditions.

(c) A study of operating experience was performed to obtain up-to-date information on iodine spiking.

Although the MB-2 program was not pursued to the full extent of the original intent (which would have concluded with verification of thermal/hydraulic codes), it did provide sufficient data to enable bounding estimates of the primary coolant entrainment in the steam released to the environment following an SGTR accident. Such bounding calculations showed that the iodine partitioning is not a very significant parameter. Because the operating experience contained only a small number of actual SGTR events, iodine spiking data from reactor trips were also included in the estimation of the primary coolant iodine levels for the radiological assessments; as a result, caution should be exercised in applying the data to steam generator accidents which include Main Steam Line Breaks (MSLB). The results of the research are summarized in Table 2 of the enclosed report.

The SRP Section 15.6.3 does not provide for a separate review procedure for the once-through steam generator characteristic of Babcock & Wilcox plants. However, because of the significant differences in the behavior of these plants from the majority of PWRs, the recommendations to resolve this generic issue are not considered to be directly applicable to B&W plants.

Recommendations

As a result of recent steam generator operating experience, NRR is undertaking a comprehensive review of the licensing basis (including operating, inspection and repair requirements) of the steam generators in the current generation of PWRs. A significant step in this direction has been publication of the draft report NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes." In this report, sample calculations are provided to explore the results of introducing more realistic parameters in the SRP calculation procedure for radiological dose assessments. It is concluded that consideration of off-site doses for design-basis events is limiting in evaluating potential primary-to-secondary leak rates. The work proceeding in this area will likely produce new staff positions regarding methods to compute radiological doses. The results of the effort to resolve GI 67.5.1, as shown in Table 2 of the enclosed report does not address or provide technical justification for changes beyond those documented in NUREG-1477.

In summary, we recommend that GI 67.5.1 be closed out with the forwarding to NRR of the recommendations in the enclosed report. Any outstanding questions, such as transient phenomena within the steam generators, are best addressed in connection with the rulemaking work wherein NRR has the lead role while RES is providing full support as requested.

ORIGINAL SIGNED BY

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

Enclosure: As stated

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REASSESSMENT OF RADIOLOGICAL CONSEQUENCES

U.S. Nuclear Regulatory Commission

Prepared by

Joram Hopenfeld

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SUMMARY

Generic Issue 67.5.1, "Reassessment of Radiological Consequences," addresses the validity of present techniques to calculate offsite radioactive dose releases from a design basis steam generator tube rupture (SGTR). This analysis provides a technical assessment of the key parameters which are used to calculate doses as described in the Standard Review Plan (SRP).

Recommendations are made to revise the SRP in light of experimental data and field experience; however, this revision will be stayed and ultimately considered with ongoing steam generator rulemaking activities. It should be noted that the recommendations made in this report do not apply to (1) SGTR accidents which include main steam line breaks, and (2) the once through steam generator characteristics of Babcock & Wilcox plants.

1.0 INTRODUCTION

Section 15.6.3 of the Standard Review Plan (SRP) provides guidance for calculating radioactivity release to the environment from a design basis steam generator tube rupture accident (SGTR). The SRP calculational model employs simplified assumptions which, in the absence of a detailed model, were considered conservative when the SRP was issued. Since that time, however, considerable information has been obtained and several SGTR accidents have occurred at U.S. plants. This new information indicates that some of the SRP assumptions are overly conservative, while others may not be conservative enough (see Table 2). The objective of this report is to summarize the latest findings regarding the SRP assumptions and recommend changes when appropriate. The experimental data were obtained primarily on recirculation type of steam generators.

In normal operation, PWR steam generator tubing can be damaged by a variety of corrosion or mechanical wear processes. If such damage causes a tube leak and if steam and water are vented to the atmosphere from the secondary cooling system, a pathway will exist for the direct release of radioactive fission products from the primary system to the environment. This possibility has long been recognized to the extent that a double-ended guillotine break of a single steam generator tube is considered as a design basis accident. Primary coolant radionuclide concentrations are limited by technical specifications so that the activity release will not exceed the dose guidelines of 10 CFR 100 (1). None of the SGTR events that have occurred in U.S. plants have exceeded these guidelines (2, 3). Nevertheless, as a result of occurrence of several SGTR accidents the staff has identified a number of issues which are related to SGTRs and described in NUREG 0844 (4). The January 1982 SGTR event at the R. E. Ginna plant, specifically, raised questions concerning the validity of a number of the SRP assumptions regarding dose calculations, (3). These concerns were discussed and prioritized in NUREG 0933 as Licensing Issues 67.5.1 - Reassessment of Radiological Consequences and 67.5.2 - Reevaluation of SGTR Design Basis.

2.0 RECENT WORK

In essence, three significant new sources of information have been produced since the SRP was issued: (1) Data from the MB-2 Steam Generator Transient Response Program, (2) laboratory and plant data on iodine partition coefficients under prototypic conditions, and (3) plant data on iodine spiking which have been collected for over a decade. No data specific to once-through steam generators are available. These new sources, which provide the basis for this report are briefly summarized below:

2.1 MB-2 STEAM GENERATOR TRANSIENT RESPONSE PROGRAM

The (MB-2) data (5) was obtained in a 0.8-percent power-scaled model of the Westinghouse Model F steam generator. It was designed to be geometrically and thermal-hydraulically similar to the Model F in important areas. At 100 percent power (6.67 MWt) it produced dry saturated steam at 6.9 Mpa (1000 psia), the same as in the Model F. The test program consisted of sixteen separate tests designed to cover a range of steady-state and transient fault conditions. These included a steam generator tube rupture (SGTR) test with a stuck open safety valve, two SGTR overfill tests, and ten steady-state SGTR tests at water levels ranging from very low levels in the bundle up to those when the dryer was flooded. Three moisture carryover tests without an SGTR were also conducted. The influence of break location and the effect of bypassing the dryer were also studied.

2.2 IODINE SPECIATION AND PARTITIONING

ORNL (6) has conducted experimental studies of iodine speciation and partitioning. The experimental system consisted of a large, 152 cm. long, 8.9 cm. diameter, stainless steel autoclave, which was heated electrically and was connected to a separate condenser vessel. Partition coefficients were obtained by sampling the liquid in the main vessel and the condenser for radioactive tracers of iodine. The results were presented in terms of pH and oxygen content in the atmosphere. Data from these tests were also compared to data on iodine speciation obtained at two power plants.

2.3 IODINE SPIKING

The maximum iodine concentration in reactor coolant, during 144 separate iodine spiking events in commercial PWRs, has been documented in LERs. All these concentrations were in excess of 1 μCi/g of I-131 and covered the period 1970-1988.

The data suggests that a trend exists towards smaller iodine spikes as more experience is obtained. This may be an indication that fuel rod manufacturing techniques are improving to minimize fuel rod cladding defects.

3.0 OBJECTIVES

In the light of the MB-2 findings and the iodine spiking study briefly discussed above it became clear that a reassessment of the present methodology in the SRP for calculating radiological consequences following steam generator tube rupture is needed. More specifically this consists of the following tasks:

1. Reevaluate the validity of using an iodine partition coefficient of 100 as specified in Section 15.6.3-4 of the SRP (1).

2. Evaluate the validity of the following two assumptions regarding initial iodine concentrations in the primary coolant:

   (a) Using the values specified by the standard technical specification (60 to 275 μCi/g depending on the reactor power) prior to SGTR initiation.

   (b) Using a value which results from an accident-initiated iodine spike (an SGTR with a coincident iodine spike) which increases the iodine release rate to a value 500 times greater than the steady-state release rate corresponding to the iodine concentration at the equilibrium value stated in the NSSS vendor standard technical specifications or plant specific technical specifications.

3. Evaluate the iodine transport model formulated by Postma and Tam in NUREG-0409. The SRP suggests that this model be used to determine iodine release. (The model embodies the concept that some of the iodine emerging from the ruptured tube may enter the steam line via liquid droplets which are entrained by the rising steam).

4.0 TECHNICAL FINDINGS

4.1 DOSE CALCULATIONS

Section 11. 10 CFR Part 100. Reactor Site Criteria requires that an individual located at any point outside the exclusion area. would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure. for two hours immediately following onset of the postulated fission product release.

The above mentioned radiation dose at the site boundary is governed by four parameters. i.e:

DOSE = (B.R) (Li) ($\underline{X/Q}$) (D.C.F)

B.R = breathing rate (meter ^{3}/sec)
Li = leakage factor (Ci. corresponding to iodine radioactivity in secondary coolant released to the environment from the safety relief valve or other mechanics)
X/Q = dispersion factor (sec/meter ^{3})
D.C.F = dose conversion factor (rem/Ci)

Of these four parameters the leakage factor Li. by far. is the most difficult parameter to calculate. and therefore it is discussed in detail in Reference 7. The difficulty stems from the dependence of Li on complex. not well understood. physical and chemical processes which occur simultaneously in the primary and secondary coolants following an SGTR event. The following mathematical formulation aids in describing these processes.

$$(1) \quad Li = \int (MC)\, dt$$

M is the mass of fluid entering the safety relief valve and ejectors following an SGTR event and C is the iodine concentration in the fluid. The term MC can be expressed in terms of its individual components:

$$(2) \quad MC = M_g C_g + M_l C_l + M_{l.b}\, C_{l.b}$$

Where.

C_g = iodine concentration in steam

C_l = iodine concentration in liquid which was entrained by steam after mixing between break flow and secondary coolant.

$C_{l.b}$ = iodine concentration in reactor coolant (RCS)

M_g = mass flow of steam

M_l = mass flow of entrained liquid from the mixed pool.

$M_{l.b}$ = mass flow of liquid which bypassed mixing.

Iodine concentration in steam is governed by a partition coefficient P. which represents the concentrations of iodine in water and steam when the two coexist in equilibrium:

$$(3) \quad P = C_l/C_g$$

The concentration C_l is determined by mixing a fraction of the break flow with steam generator inventory. The iodine concentration in the liquid at the break site is the same as the primary coolant iodine prior to the accident multiplied by the iodine spike S. i.e:

$$(4) \quad C_{l.b} = S\, C_o$$

S = an iodine spike factor which increases the initial iodine concentration C_0. during the SGTR event. (The iodine spike results from large changes in reactor power or RCS pressure transient)

Once the entrainment M_1. the bypass flow M_{1-b}. the partition coefficient P. and the iodine spiking factor S are specified. the above equation can be evaluated with the aid of available thermal hydraulic codes such as RELAP Reference 8. The partition coefficient. P. pool entrainment M_1. coolant bypass M_{1-b}. and iodine spiking are each discussed below in accordance with the objectives.

(a) <u>Partition Coefficient - P</u>

Section 15.6.3-4 of the SRP specifies that "an iodine partition coefficient of 100 between steam generator water and steam phases may be conservatively assumed." Review of the literature on iodine partition coefficient P. however. indicates that the ORNL data (<u>6</u>) is the most applicable data for the determination of P. All other published data were obtained under conditions which are nonprototypic with respect to iodine concentrations. In operating reactors. the iodine concentration is on the order of 10^{-10} Moles per liter while the published data on the partition coefficient are all based on concentrations in the 10^{-5} Mole per liter range. However. during and immediately following iodine spikes in range of 12 μCi/g (as proposed). the iodine concentration will be closer to 10^{-5} Moles/liter range.

Table I. shows that the partition coefficient strongly depends on pH (the higher pH will result in higher iodine partition coefficients). During normal operation. the pH in PWR's is in the 6.0 - 9.5 range on the primary side. and about 9.5 on the secondary side. The introduction of boron on shutdown. however. reduces the pH on the secondary side. EG&G performed a study to determine the secondary system pH following an SGTR for a wide range of PWR designs (10). In general. it was determined that the pH decreases from an initial value of approximately 8-9 to a value of approximately 6.5. independent of PWR design.

e noted that the partition coefficient as reported in the is sometimes based on concentrations by mass and sometimes on convert P from one base to another, a multiplication by the io is required:

$$(5) \quad P \text{ (mass basis)} = P \text{ (Volume basis)} \times (Dg/Dl)$$

Dl and Dg are the densities of the liquid and the gas respectively.

ENTRAINMENT - M_1

el paths exist (7) for the release of water droplets containing o the steam space: (1) Upon entering the secondary side of the rator, the flow from the ruptured tube partly flashes into steam and atomized into small droplets. These small droplets may be captured bubbles and thus enter the steam space. (2) Following complete tween the primary break flow and the secondary side mass of liquid. plets are entrained by steam at the pool surface due to the ordinary rocess. (Liquid ligaments, originating from bubbles as they burst at steam interface, are broken into small droplets which in turn are y the rising steam).

flow of small water droplets thus introduced into the steam is referred to as the moisture carryover. To avoid damage to the moisture carryover must be kept at a minimum during plant operation. ilibrium carryover can be calculated reliably. the carryover during t operation may be different because of level swell. The SRP, Section item 10, by reference, recommends that the model described by Postma (10) be used to describe iodine transport. This reference gives the g equation for iodine concentration in steam due to pool entrainment.

$$(6) \quad C_s = 2.5 \times 10^{-3}\, C_b$$

Where,

C_s = concentration of iodine in steam due to boiling carryover
C_b = iodine concentration in boiler water

The MB-2 experiments (5) simulating a tube rupture in a Westinghouse model F steam generator indicate that equation 6 would overestimate entrainment by at least three orders of magnitudes. (page iv. of Reference 5).

4.4 PRIMARY COOLANT BYPASSING - $M_{1.b}$

Postma & Tam (10) postulated that the major mechanism for liquid entrainment in steam is the scrubbing of atomized droplets at the break site by rising steam bubbles. These bubbles break at the water steam interfaces and allow small droplets to be carried by the rising steam. Postma & Tan then derived the following equation for this process:

$$A = \exp\,(1.724 \times 10^{-5}\, d^2 h) \qquad (7)$$

Where:

d = diameter of water droplet, microns
h = height of water over break, cm
A = Attenuation of liquid droplets.

The experimental data generated in the MB-2 program shows that very little or no primary coolant bypassing exists for a wide range of operating conditions. In the few instances where bypassing was observed, it was on the order of 0.001% and occurred only at bottom break locations. These results are in disagreement with Equation 7 which predicts an increase in iodine release as the height above the break is reduced

The report, Reference 5, which summarized the MB-2 program, presented (page 7-11) a discussion of flashing, atomization and bypass. Nevertheless, since these terms have been misunderstood, an expanded discussion of these parameters follows

The pressure and temperature differences between the primary side and the secondary side of the reactor system. dictates that a portion of the break fluid flash into steam. The fraction of the liquid which flashes into steam is calculated by equating the enthalpies across the steam generator tube at the break point. Accompanying flashing. another process. the atomization of the liquid into small droplets. also takes place. This process is much less understood than flashing and the droplet size. can at best. be estimated only approximately within an order of magnitude.

When the steam bubbles break at the liquid level the fine droplets are released and swept by the rising steam from the pool. Since these droplets. did not mix with the bulk secondary fluid they have been defined in Reference 5 as bypass flow. Prior to the MB-2 program it was believed that because of their small size these droplets could present an effective vehicle for the release of iodine to the environment. No theorical basis is available to estimate the mass of liquid that may reach the SORV. However. the MB-2 tests offers an experimental basis for making such an estimate. Since radiological releases due to iodine volatility can be conservatively bounded it was not necessary to incorporate iodine in these tests. It should be noted that the bypass data reported in Reference 5. applies only to the iodine which would be carried by the liquid droplets during the actual SGTR transient.

The iodine carried by the fraction of the liquid which flashed into steam is also a source of radiological release to the environment. The process which controls this source of iodine as it reaches the SORV. is distinctly different than the bypass process.

Because the time scale for flashing is on the order of milliseconds it is appropriate to assume that the steam will contain all the iodine which previously was concentrated in the equivalent mass of liquid. As the steam flows through the steam generator. chemical kinetics dictate that some of that iodine be transported to the surrounding bulk liquid as well as the entrained droplets. The rate of such transport is controlled by fairly well understood gas/liquid mass transfer phenomena and can be predicted by RELAP if the pressure, temperature, liquid level, and bulk iodine concentration and the equations which describe iodine oxidation and hydrolysis are specified. When tube rupture occurs near or at the water level such detailed calculations are not required. In this limiting case, one may consider the flashing steam as equivalent to bypass by atomization, because in both cases the iodine which emerged from the break site was prevented from mixing with the secondary fluid. From practical considerations, since the break location is not known, it is not necessary to model the iodine in the flashing steam; it is sufficient to assume that all the iodine in the steam will reach the SORV without retention in the steam generator. Because of plate-out and contacts with the liquid in the upper deck, some of the iodine will not reach the SORV and therefore such calculations should be regarded as conservative.

In summary, experimental data for W type steam generators show that iodine bypass is very low. The unpredicable break location in an SGTR together with poorly understood iodine chemistry preclude sophisticated modeling of iodine during the SGTR transient. Bounding, but conservative, considerations are adequate to insure public safety. On the other hand, if one is interested in a more detail study of iodine transport it is necessary to specify the temperature, the pressure, the mixing, and the pH in the secondary volume as a function of time. The RELAP code, together with the appropriate chemical equations could provide such details. The derivation of this information is beyond the scope of this study.

The RELAP code is also required for the bounding calculations in order to determine the primary to secondary leakage, the steaming rate and the frequency and timing of the SORV opening. Given the many simplified assumptions which are inherent in these calculations, an exact numerical value for the partition coefficient is not required and approximation in the range of an order of magnitude is adequate.

4.5 IODINE SPIKING, S

(1) Iodine Spike Prior to SGTR Initiation

As mentioned above, the SRP requires that the licensee assume that the iodine spike raised the concentration of iodine in the primary coolant to a constant value which is specified by the Standard Technical Specifications. This value ranges from 60 to 275 μCi/g depending on reactor power as shown in Figure 1.

EG&G conducted a statistical analysis of the available plant data to assess the above iodine concentrations in the light of recent plant experience. In this analysis (11) it was assumed that the events represent a random sampling of the iodine spiking which has occurred and is expected to occur in commercial PWRs. No attempt was made to correlate the data to either specific plants or fuel manufacturers and the results are independent of any assumption regarding the shape of the probability distributions which are measures of the probability that any random iodine spiking event would result in a magnitude (either maximum iodine concentration or iodine release rate) less than a given value. Both the nominal probability distribution and the 95% confidence limit probability distribution were calculated using binary distribution statistical analysis methods.

The cumulative probability distributions for the maximum reactor coolant iodine concentration is shown in Figure 2. A comparison of this figure with Figure 1 clearly indicates that the iodine spike as presently used in analyzing SGTR events is too conservative by at least a factor of three. The maximum value in Figure 2 is 19 μCi/gr while the minimum value in Figure 1 is 60 μCi/gr.

(2) An SGTR with a Coincident Iodine Spike

EG&G (11) has collected and analyzed radiochemistry data from 26 PWRs following reactor trips and presented the results in terms of a probability that an SGTR will not exceed a certain iodine release rate. These results clearly indicate that the SRP methodology of using 1 μCi/g as the initial iodine concentration and a 500 fold increase in release rate is overly conservative by at least an order of magnitude.

5.0 DISCUSSION OF TECHNICAL FINDINGS

5.1 IODINE SPIKE

Based on the evaluation reported in References 11 and 12, it is recommended that (1) equilibrium iodine concentration associated with a preaccident iodine spike be changed to 12 μCi/g, and (2) the iodine release rate from fuel rods to the primary coolant coincident with a SGTR event be changed to 1.35 Ci/h-Mw(e).

5.2 PARTITION COEFFICIENT

The results of Table 1 are the only available data on iodine partition coefficient at concentrations which represent those exist in the RCS during an SGTR event. The main purpose of the experimental program from which this data was derived was not to obtain precise values for the partition coefficient but rather to resolve a large uncertainty in the published values of the partition coefficient.

Reference 13 indicates that the pH in the secondary system following an SGTR decreased to a value of about 6.5 in approximately 10 minutes. Under these conditions, best estimate value for iodine partition coefficient would be 35 (mass basis) compared to 100 given in the SRP. The higher partition coefficient will result in lower radiological consequences.

The sensitivity of the dose release to the partition coefficient depends on whether the major mass of the steam at the SORV originated from the pool or from the flashing steam at the break site. As discussed in Section 4.4 above, a thermal hydraulic computer code (RELAP) is required to perform such calculations. Generally speaking, during the initial part of the transient, the dose will be only slightly dependent on the partition coefficient. Further into the transient, pool steaming will predominate and the dose release would become more directly dependent on the partition coefficient as long as the moisture carryover is as low as indicated by the MB-2 tests

5.3 ENTRAINMENT

The experimental data on moisture carryover from the MB-2 tests (5) show that the present practice of using equations 5 and 6 to describe iodine transport would yield doses which are overly conservative. There also appears to exist an inconsistency between the theoretical model in NUREG-0409 for bypass entrainment and the MB-2 data.

The lack of contribution of entrainment to total iodine transport is illustrated by comparing iodine transfer by entrainment to transfer by volatility. The total amount of iodine retained in the water inside the steam generator vessel divided by the iodine present in the main steam line at any instant is commonly referred to as the decontamination coefficient, DF. If P is the partition coefficient of iodine and E is the moisture content of the steam, then by this definition we have:

$$1/DF = \frac{\text{Iodine In Steam Line}}{\text{Iodine In Steam Generator Vessel}}$$

$$= \frac{\text{Iodine in Vapor + iodine in liquid}}{\text{Iodine in liquid in vessel}} = 1/P + 1/E$$

Since P ranges from 17 to 2300 and E is on the order of 10^6, it is clear that iodine transport by entrainment can be neglected. The SRP guideline (15.6.3-4, item 10) of using the model described in Reference (10) appears to be at variance with the experimental data.

The above considerations are not applicable to once through steam generators. The degree of entrainment in these units depends highly on break location. Breaks near the upper tubesheet could provide primary flow a direct path into the steam line. However, since no data is available to calculate what fraction of the flow at the break site will enter the main steam line

In conclusion, although many complex considerations were included in this evaluation of the methodology to estimate the off-site dose from SGTR the final results appear to be affected only to a minor extent. The reduction in the magnitude of the iodine spike together with the elimination of iodine transport by droplet entrainment may off-set the increase in dose releases due to a lower partition coefficient.

6.0 CONCLUSIONS

A steam generator tube rupture has been observed to be relatively a high frequency event, (10^{-3} to 10^{-2} per reactor year) and therefore, it is expected that some amount of iodine will be released to the environment as a result of an SGTR accident. Therefore, it would be prudent to include in the SRP the latest information.

The information dictates that the iodine spike and partition coefficient be reduced as discussed and the bypass be eliminated. The recommend changes in the SRP are indicated in Table 2.

7.0 RECOMMENDATIONS

Specific changes are as follows

PAGE 15.6.3 3

Item 6 (a). Substitute

A reactor transient has occurred prior to the postulated steam generator tube failure accident and has raised the primary coolant iodine concentration to 1. µCi/g

PAGE 15.6.3-4

Item 6 (b). Substitute:

The reactor trip on the primary system depressurization associated with the postulated accident creates a iodine spike in the primary system. The increasing primary coolant iodine concentration is estimated using a spiking model which assumes that the iodine release rate from the fuel rods to the primary coolant is 1.33 Ci/hr per NW(e).

Page 16.6.3-5

Item 10. Substitute:

Determination of the iodine transport to the atmosphere. For circulating type steam generators. a fraction of the iodine in the primary coolant escaping to the secondary system is assumed to become airborne immediately due to flashing and atomization. A value 0.001% should be assumed for the atomization fraction. That fraction of the primary coolant iodine which is not assumed to become airborne immediately enters the secondary system water and is assumed to become airborne at a rate determined by the steaming rate and iodine partition coefficient. A value of 0.005% should be assumed for the moisture carryover. An iodine partition coefficient of 35 (mass basis) may be assumed between the steam generator water and steam in recirculating type steam generators.

Table 1. Summary of results from steam generator iodine experiments 285°C, 1000 psi, 0.2 M borate, 1.0E-9 M I (Reference 6)

Atm	pH at 25°F	I_2 in liquid (%)	Organic I in liquid (%)	Partition coefficient PC	
				(Volume)	(Mass)
Argon	5	2.04	0.11	6.87E+03	333
	7	0.44	0.07	5.18E+03	251
	9	0.02	0.00	4.75E+04	2300
Air	5		3.95	3.50E+02	17
	7	1.20	0.15	8.88E+02	44
	9	0.12	0.01	7.16E+03	360

	Current SRP Guidelines	Proposed Change to SRP
Partition Coefficient	100 No Basis given (Mass or Volume)	35 (MASS BASIS)
Pool Entrainment (Recirculating Type)	Equation 27. Ref. 8	0.005%
Bypass Entrainment	Equation 32. Ref. 8	0.001%
(a) SGTR Following Iodine Spike	Iodine Concentration In RCS 60 - 275 μCi/g	12 μCi/g
(b) SGTR with A Coincident Iodine Spike	500 Increase in release rate (Initial Concentration = 1 μCi/g	1.33 $\frac{Ci}{hr*MW(e)}$ (Initial Concentration 1 μCi/g

TABLE 2 - SUMMARY OF RESULTS

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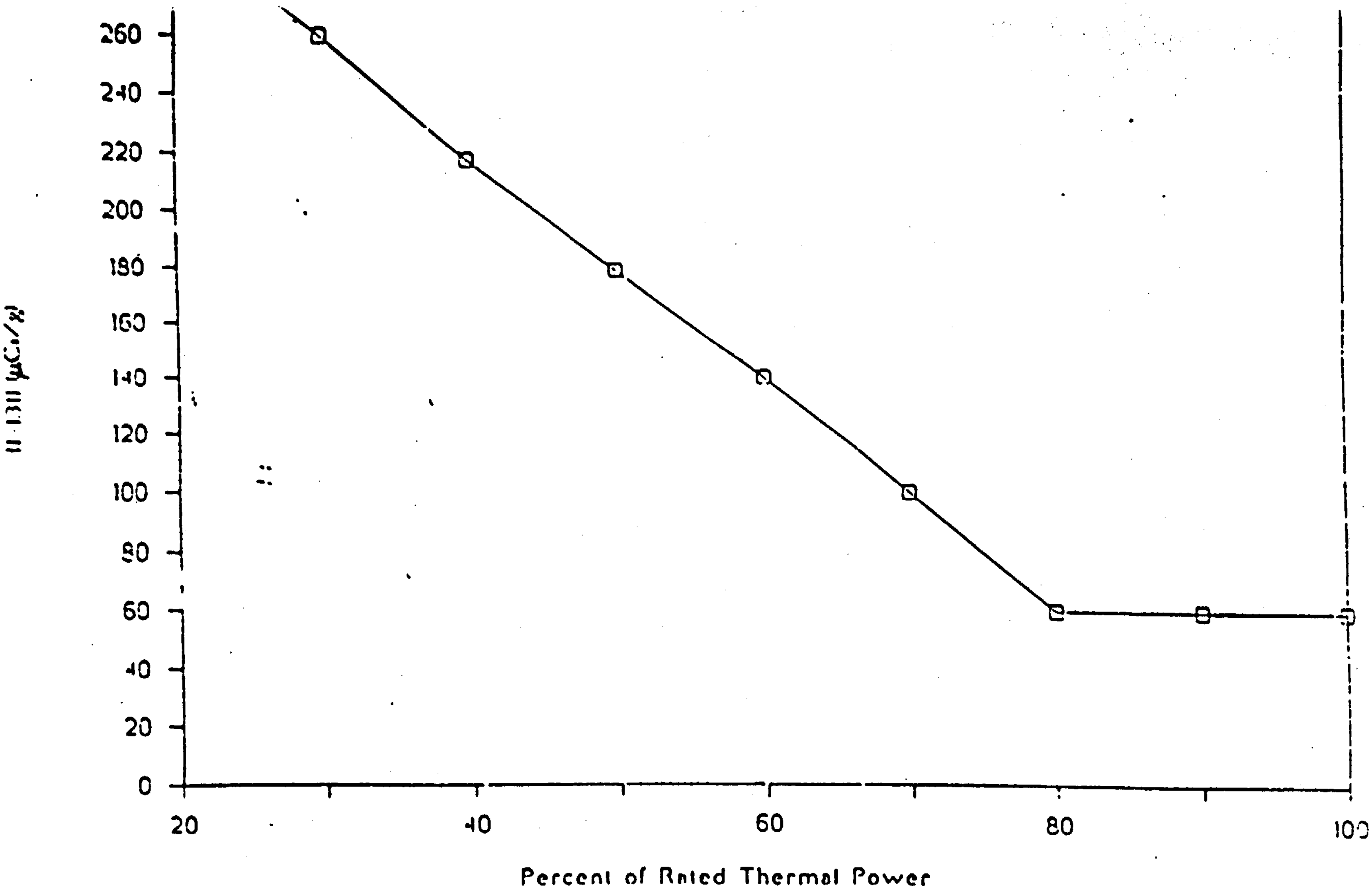

Figure 1: SRP specified iodine concentration for Category 1 SGTRs

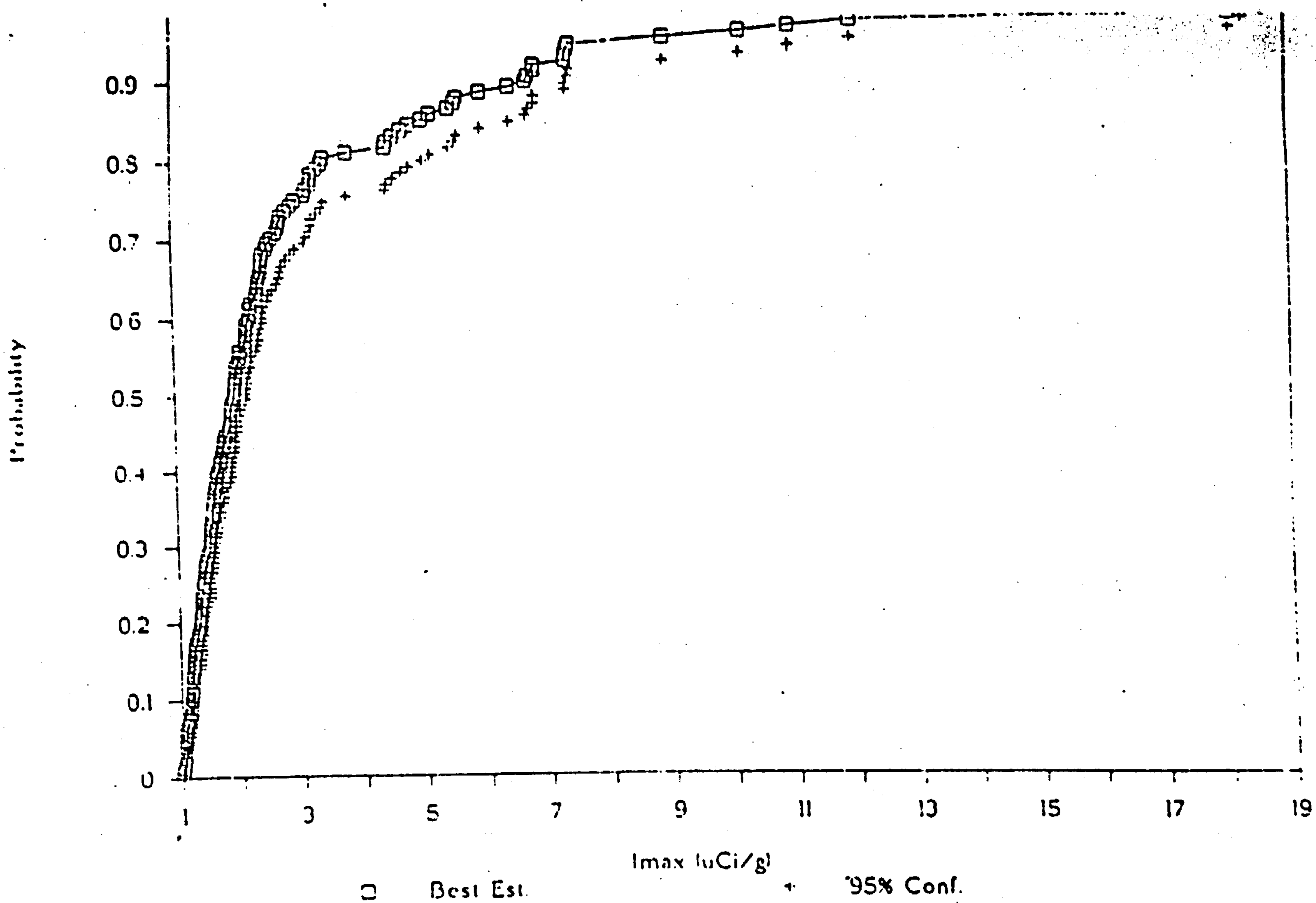

Figure 2: Probability of maximum iodine concentration