ATTACHMENT 8

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

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MEMORANDUM FOR: James M. Taylor
Executive Director For Operations

FROM: Joram Hopenfeld
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Office of Nuclear Regulatory Research

SUBJECT: DIFFERING PROFESSIONAL OPINION REGARDING VOLTAGE-BASED INTERIM REPAIR CRITERIA FOR STEAM GENERATOR TUBES

This is a differing professional opinion (DPO) concerning a newly proposed NRC regulation (1) allowing Westinghouse plants to operate with degraded steam generator tubes. The key issue raised by the proposed draft Generic Letter (GL) is whether the new steam generator tube repair criteria introduce risks which significantly lower existing margins of health, safety and environmental protection.

In my opinion, core melt frequency will be increased to 3.4 E-4, and 10 CFR 100 dose limits will be exceeded if the GL is approved.

My concerns were raised in a DPV dated December 1991, but were not resolved to my satisfaction. This DPO is in accordance with established DPV/DPO procedures, Directive 6.2.

BASIS FOR CONCERN

PWRs were licensed on the basis that they would not increase public risk if a main steam line break (MSLB) were to occur. The operating license of these plants require that steam generator (SG) tubes be plugged when wall degradation exceeds 40%. However, the design and material selection are such that the entire wall thicknesses of SG tubes, mostly at support plate locations, become increasingly populated with through-the-wall cracks. Westinghouse claims that even though the units were not designed to prevent cracked tubes from leaking they still will not leak significantly. This theory cannot be tested against field experience; however, the support plate/tube structure is not a leak tight joint. The crux of the issue is how much the tubes will leak during design basis accidents.

When deteriorating SG units are subject to MSLB loads, the primary radioactive coolant can directly escape to the environment through the cracked tubes and the openings in the steam pipe. The resultant contamination depends on the amount of primary/secondary leakage, the amount of coolant activity and the location of the MSLB. If the leakage is on the order of 600 gpm and the MSLB occurs upstream of the isolation valve the release could be very large. If the leakage is in the 10-100 gpm range, contamination will be relatively small but sufficient to exceed dose limits as specified in 10 CFR 100.

Large Leakage

PWRs are designed to reuse lost coolant following pipe ruptures as long as the rupture occurs within containment. When the SG operates with through-the-wall cracks and a steam line break occurs outside containment, the plant is not capable of reusing the escaping coolant. Unless the affected unit can be isolated and the accident terminated the core will eventually melt with an open path for highly radioactive fission products to contaminate the environment. Because of this bypass feature such accidents have been discussed in NUREG-0844; however, because of the single failure criterion, they were considered hypothetical. The purpose of the 1991 DPV was to point out that with through-the-wall cracks the possibility of containment bypass can not be treated as an academic, what if, paper study.

Using a mechanistic approach NRC/RES (2) predicted leakage of 33-1350 gpm. NUREG 1477 proposes that the leakage can not exceed 1000gpm because of ECCS pump capacity limitations.

The above leakages can lead to a core melt and are best discussed in terms of risk which varies considerably. NUREG-1477 predicts a core damage frequency of 6.3 E-7 while NRC-RES (3) predicts 3.4 E-4. I believe that the latter is a more realistic estimate because of problems with the NUREG leakage model.

Small Leakage

Assessment of whether 10 CFR 100 limits will be exceeded depends on the accuracy of leakage and dose predictions.

Leakage Calculations rely on statistical correlation from laboratory tests using specimens which were either conditioned in the laboratory or cut from pulled tubes. The statistical approach involves many assumptions:

- Eddy-current voltage signals are a measure of leakage.
- Growth rate and coalescence of cracks can be simulated in the laboratory.
- A network of axial cracks will not exhibit circumferential crack characteristics, and
- Cracks with long incubation period and subsequent fast crack growth rates will be detected in time.

Recent field experience (Palo-Verde, Summer and Braidwood) shows that undetected cracks can grow very fast. When such high growth rates and outliers are excluded, the statistical correlation would predict low leakages, and therefore reduce tube plugging while exhibiting that 10 CFR 100 limits are met.

The predictions in the GL are based on data most of which was generated by Westinghouse. The leakage predictions can vary widely and provide ns considerable freedom for continuous manipulation of the voltage plugging criteria. The stimulus for the development of the statistical approach was the realization that crack formation and leakage were too complex for analysis.

Now, however, Westinghouse proposes to eliminate outliers by exactly such analysis. At the NRC/Westinghouse meeting it was apparent that the NRC staff is only superficially familiar with how Westinghouse generates the data. In my opinion, even if it were possible to continuously monitor all the thousands of cracks simultaneously in situ, it would still be impossible to predict leakage within the required accuracy. In contrast, Westinghouse/NRC staff believes that eighteen month interval inspections are sufficient to predict leakage within several gpm.

Dose Calculations depend on the initial coolant iodine activity and iodine spiking. Iodine activity limits are controlled by the technical specification (TS) of each plant. Iodine spiking occurs when the power, the temperature or the pressure are perturbed. Increases as high as 10,000X in coolant activity following reactor shutdown have been observed, (4). Present licensing for the design basis SGTR and the design basis MSLB, is based on an empirical iodine spike which corresponds to iodine release rate from the fuel to the coolant which is 500 times the equilibrium release rate. The MSLB/Leakage event can be characterized in terms of a fast reactor coolant pressure drop, (to 800 psi in 200 seconds). It is this pressure transient and the large leakage that distinguishes the MSLB/Leakage from the SGTR and MSLB. The data base for the 500 spike does not include MSLB/Leakage transients, it is expected, however, that these transients will exhibit much higher spikes. Reactor experience (5) indicates that it is the tail end of the pressure transient which most significantly affects spiking, and Figure 1 shows that a pressure change spike releases as much activity into the coolant as a power ramp spike. The present licensing requirements for off-site radiological consequences, as specified in SRP-15.6.3., must be corrected when they are applied to MSLB/Leakage accidents. It should be noted that present dose calculations for the design basis MSLB are based on total leakage of one gpm; therefore, a correction here is not very significant because of the low leakage. When, however, the leakage is large (10 gpm or more) the effect of rapid pressure changes on the spike can no longer be ignored. Not only does the GL ignore this issue, it also uses the TS to incorrectly tune dose predictions.

On September 28, 1993 NRC approved Farleys' request to lower TS limits by a factor of four to allow post-MSLB primary/secondary leakage. Similarly, in May 1994, NRC allowed Braidwood Station, Unit 1 to lower their TS from 1 to 0.35 microcurie per gram of coolant and thereby lower the predicted dose from 80 rem to 28 (30 is the SRP limit). The flaw in the above decision is shown in Figure 2 where the spike increases with reduction in the initial iodine activity. In my opinion the appropriate procedure would have been to multiply the 80 rem by a factor of 10, at minimum, to account for fast pressure transients.

DISCUSSION

The proposed GL implies that there is a known correlation between initial coolant activity and dose release, and that lowering the technical specification (TS) limits "is an acceptable means for accepting higher projected leakage rates and still meeting the applicable limits of 10 CFR 100 utilizing licensing basis assumption." Figure 2 indicates that this assumption is not valid, lower coolant activities result in higher spikes.

To mitigate Dr. Buslik's conclusion (6) that 10 CFR 100 are being exceeded, NRC-RES (7) states that 10 CFR 100 will be met because of recent improvements in fuel reliability. The rationale is that the 0.5% fuel failure used by Dr. Buslik is an order of magnitude higher than recent experience indicates. RES ignores the fact that by this reasoning a fuel failure of 0.05% will reduce the 1500 rem calculated by Dr. Buslik to 150 rem which is within 10 CFR 100 but exceeds the GL limits by 120 rems. The RES position raises several issues but because they are beyond the scope of this DPO, only one comment is provided (see footnote).

I do not agree with RES (7) that the GL represents a significant accomplishment. It is based entirely on the Westinghouse methodology and ignores data which does not support it. As a defendant in SG related legal actions and as a consultant to several utilities, Westinghouse stands to benefit considerably from the GL. Since the subject matter involves very complex technical issues, a thorough independent technical peer review of the proposed plugging criteria is required. The task group was not a substitute for a peer review because it simply endorsed the Westinghouse approach with some minor modifications.

In early 1993 an NRC task group was established to evaluate the issues which were raised by the DPV and Dr. Muscara. The groups' work was concluded with a draft NUREG-1444 which was issued for public comment in June 1993. The House Committee on Natural Resources Subcommittee on Energy and Mineral Resource was briefed accordingly. Westinghouse/EPRI/Utilities, strongly objected to the NUREG conclusions. Without new data and in spite of field reports on high crack growth rates and objections (1) from three task group members the NUREG recommendations were revised by NRR as requested by Westinghouse/Industry. Dr. Muscara is a senior materials expert with 15 years SG related experience, Dr. A. Buslik is a senior PRA expert with experience in iodine spiking problems, and I myself have 12 years 10 CFR 100 related experience. The revised NUREG conclusions were different from those which were provided to the House Committee and the ACRS earlier.

Insufficient understanding of the technical issues by the staff combined with selective presentations of data by Westinghouse resulted in changing the plugging criteria from 1V to 2V. Even though this change has not yet been approved, Westinghouse already is discussing (8) changes which would increase the plugging criteria to 3V and more.

I have raised the 10 CFR 100 concern at several NRC/EPRI/INDUSTRY meetings and in writing (9). A meeting with industry experts was scheduled tentatively by EPRI (Steininger). In a report submitted to EPRI (January 1994, TR-103680) Dr. Postma clearly points out that it is only by assumption that "the available sample of spike events is representative of the population of spikes that could occur in postulated SGTR/MSLB accident sequences." In spite of these concerns the GL ignores the entire issue.

Directive 6.2 "Differing Professional Views or Opinions" specifies that if after receiving the Office Director's report "the (DPV) submitter does not consider the matter closed, a written DPO statement expressing continuing concerns may be submitted to the Commission or EDO, as appropriate." 30 months following submittal of the DPV I still have not received a report from the Office Director. In the absence of a reply from the Office Director, the lack of any activities on GI-163 and the inadequacy of the proposed Generic Letter I consider the DPV step as unresolved and therefore, I am proceeding with the formal DPO process.

RES, in response to the DPV, requested that I provide additional information so the issue can be prioritized in accordance with RES Office Letter No 1. In September 1992, RES opened a new Generic Issue (GI-163), ranked it a HIGH priority, and sent it for peer review. Contrary to Letter No. 1 and in spite of a written reminder (10) to the Director I was excluded from this review. GI-163 - MULTIPLE STEAM GENERATOR LEAKAGE is still in Priorization (step 4).

It is disturbing that after more than 30 months since the concern has been brought to RES management attention no data has been generated to allow independent assessment of the Westinghouse approach and provide sound rationale for Commission review.

The lack of prompt disposition of safety concerns which I have brought to RES management attention has been systematic and pervasive. For example, in 1987 I have predicted certain complex SG degradations, (11). When RES failed to act I raised the issue with the Commission which promptly issued an inquiry (12). Several years later the degradation occurred almost exactly as predicted at San-Onofre (13) and Maine Yankee (14). I believe that if prompt action had been taken in 1987 unnecessary risk to the plant and costly outages could have been avoided.

In summary, public health and safety can be best protected by replacing the affected steam generator units and not by the institution of easily tunable plugging criteria.

CONCLUSIONS

My differing professional opinion primarily concerns the following:

The GL methodology will allow plants to exceed 10 CFR 100 limits. At least one plant, Braidwood, already exceeds these limits.

I disagree with the position that the Westinghouse statistical model allows conservative leakage predictions. Consequently the risk of core melt is much higher than indicated by the proposed GL.

I disagree with issuing the proposed GL for public comments without accompanying documentation which would allow an assessment of the following issues:

1. How the data on tube inspection at Palo Verde and Braidwood fit in the leakage correlation.

2. The correlation between the Technical Specification of allowable coolant activity limits and the radiological consequences.

3. The effect of fast depressurization on iodine spiking.

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REFERENCES

1. CRGR/DRAFT GL Package on Voltage-Based Repair Criteria For Steam Generator Tubes.

2. Memorandum, RES to NRR, Jan. 15, 1992.

3. Memo, Heltemes to Gillespie, GI-163 Multiple Steam Generator Tube Leakage, Sept. 1992.

4. LER-009/03L-0

5. WCAP 8637

6. Memo, A. Buslik to J. Strosnider, Comment On CRGR/DRAFT GL, June 6, 1994.

7. Memo, T. Speis to A. Thadani, Same subject, July 1, 1994.

8. Letter, N. Liparulo to B. Sheron, Westinghouse Anticipated Licensing Activity Addressing Steam Generator Tube Integrity Issues, June 27, 1994.

9. Memo, J. Hopenfeld to J. Strosnider, December, 1993.

10. Memo, J. Hopenfeld to W. Minners, "EDO Request Regarding Staff Concerns" Oct. 22, 1992.

11. Memo, J. Hopenfeld to L. Shotkin "Concern Regarding Loose Parts in W Steam Generators," March 10, 1987.

12. Memo, J. Asselstine to V. Stello "Surry Pipe Break" April, 13, 1987.

13. Information Notice, 91-19, Steam Generator Feedring Distribution Piping Damage.

14. C-E Infobulletin 92-01, "Steam Generator Component Erosion/Corrosion Discovered at Maine Yankee," April 1, 1992.

NOTE

RES methodology is based on the assumption that a correlation exists between fuel failure and actual coolant activity. Because of several reasons (legal, method of detection, etc.) the degree of reported fuel failures is not uniform throughout the industry. The reported and actual fuel defects may be related, but unless they can be described, improvements in fuel reliability are no indication that 10 CFR 100 will be met. Iodine coolant activity depends on the actual fraction of fuel rods that leak, the plenum/gap rate constant, reactor power level and the cleanup rate. Estimates of the rate constant vary over an order of magnitude indicating that iodine release is controlled by a large number of variables. Size and location of the defects are probably among them and therefore fuel failure is a very poor indicator of iodine release.

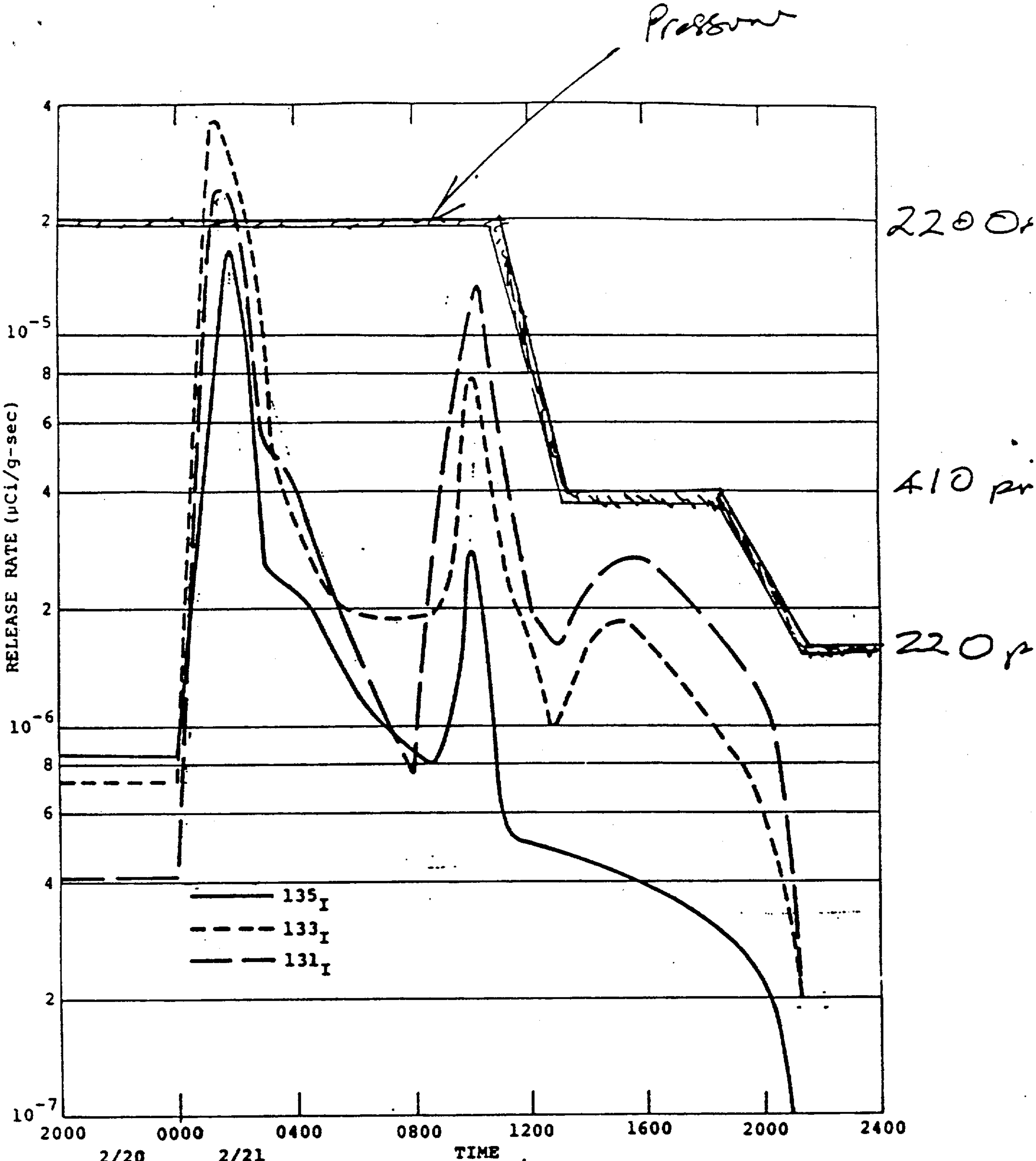

Figure 1

THREE MILE ISLAND SHUTDOWN "SPIKING" IODINE RELEASES FROM FUEL INTO PRIMARY COOLANT (μCi/g-sec) 2/20 - 2/21/76

EPRI
NP-939

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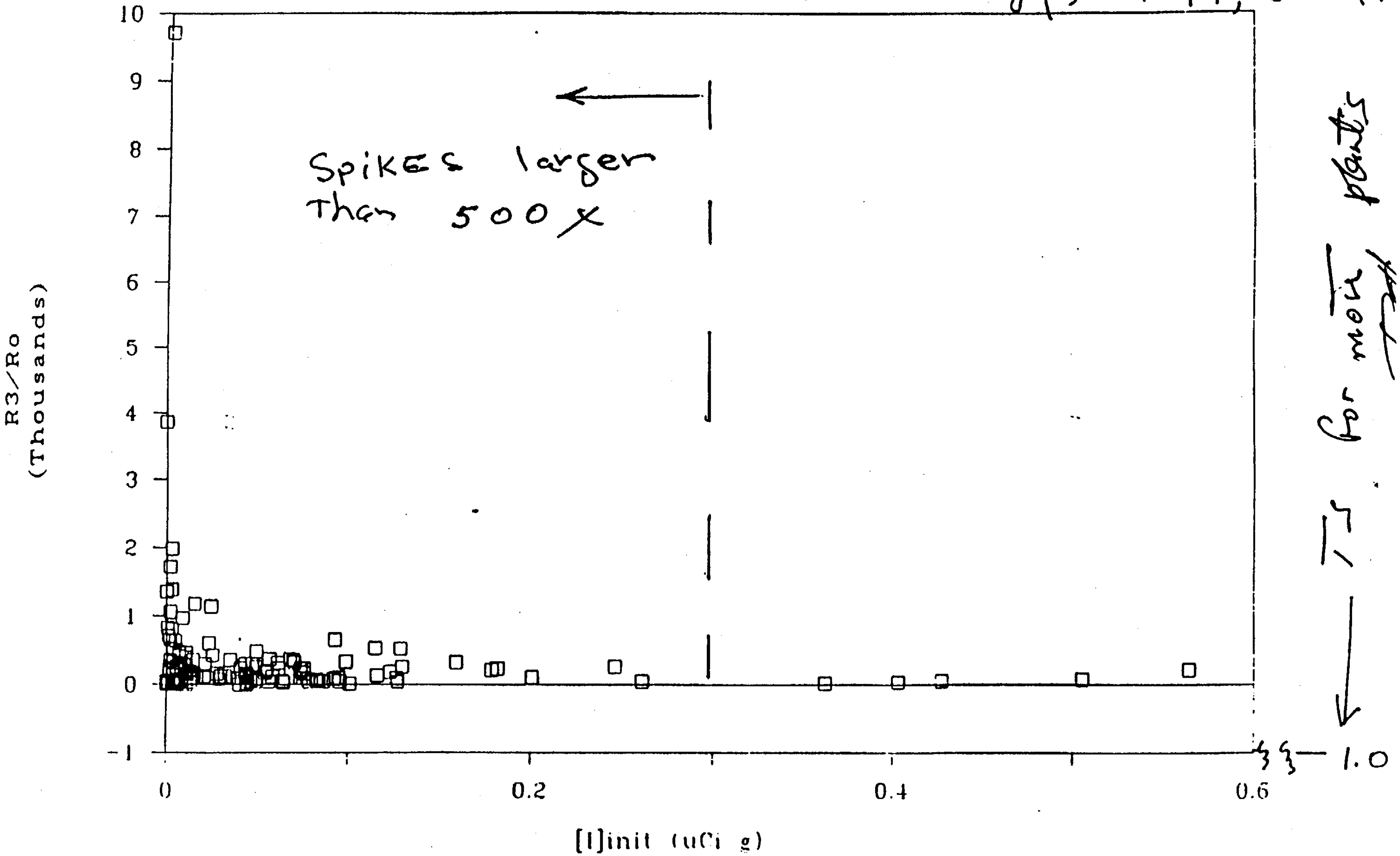

Figure 2 Release rate ratio (R3/Ro) versus the initial iodine concentration