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

A. Thadani
~~W. Kniel~~
File 5.1

SEP 8 [illegible]

V. Stello, Assistant Director for Reactor Safety

WESTINGHOUSE ATWS CATEGORY B - LOSS OF NORMAL AC POWER

Attached is a brief writeup, tables, and figures presenting the results of our preliminary review of the loss of AC power event with a failure to scram for Westinghouse ATWS Category B plants. Based on this preliminary review, we conclude that the containment will have to be isolated on some signal prior to releases through the pressurizer to avoid doses substantially in excess of 10 CFR Part 100. With respect to leakage through the secondary side, dose consequences approach 10 CFR Part 100 limits if a substantial amount of fuel is failed or if steam generator leakage increases to several gallons per minute as a result of the transient. These doses are for a X/Q of 10^{-3} sec/m^3. The meteorology at some sites could result in doses which are higher than those presented here by a factor of three or four. In addition, we would like to emphasize that the failed fuel was assumed to contribute only fuel gap activity to the source term. No additional releases from the fuel itself were considered. We would need time/temperature data for the fuel to consider this source.

We would like to reiterate that these are preliminary calculations based on the rough data given to us by your staff. Once the final numbers for the primary and secondary systems' performance, failed fuel fractions, etc. are established, we will have to redo our calculations. For instance, we do not feel that the ten-minute delay for operator reaction to the event and manual actuation of the safety injection has been properly justified. We also want to note that we have made the non-conservative assumption that both centrifugal charging pumps are available for safety injection at the request of your staff. (We note that WASH-1270 calls for a "conservative" rather than a "realistic" evaluation for Category B plants.)

Any questions on the calculations or assumptions used should be directed to E. Adensam or H. Fontecilla, Accident Analysis Branch.

Brian K. Grimes

Brian K. Grimes, Chief
Accident Analysis Branch
Division of Technical Review
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See page 2

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cc: R. Heineman
F. Schroeder
H. Denton
T. Novak
W. Minners
A. Thadani
L. Olshan
W. Pasedag
E. Adensam
H. Fontecilla

WESTINGHOUSE ATWS
LOSS OF NORMAL AC POWER

The loss of normal AC power with failure to scram has been analyzed for potential radiological consequences using the assumptions presented in Table 1. Consequences from steam generator leakage and from containment leakage have been estimated. Figures 1, 2 and 3 present the estimated thyroid dose from two hours of release from the steam generator relief valves. This was calculated assuming an I-131 equivalent source term from the failed fuel and in the coolant. A partition factor of ten is normally assumed for the primary coolant fraction that flashes to steam when leaked into the secondary side of the steam generators. However, for these estimates that factor was reduced to 1.0 for the one to thirty minute time period because the secondary side water volume was not sufficient to cover the tubes during that time period. (This assumption increases the computed doses by about 30%.)

The dose estimates were made using an I-131 equivalent source term and a two-hour X/Q of 10^{-3} sec/m^3. An iodine spike was assumed to occur in the time period following the transient, but this did not contribute significantly to the doses for the cases where fuel failure was assumed to occur.

The containment leakage source term was the primary coolant mass leaked into the containment through the pressurizer relief valves as a function of time. A 2.5×10^6 ft^3 containment volume with a 0.1%/day leak rate was assumed. Complete mixing in the primary coolant and in the containment was also assumed. Of the iodines released to the containment, 50% were assumed to be in the primary coolant flashed to steam and 50% remained with the liquid and were not considered available for release.

Table 2 presents the estimated containment leakage doses. These estimates assume the containment isolates prior to any releases from the pressurizer. Currently there has not been a mechanism identified which assures containment isolation prior to 10 minutes. Without isolating the containment and assuming the releases are directly to atmosphere, the estimated doses are far in excess of 10 CFR Part 100 exposure guidelines.

Whole body dose estimates were also made for the steam generator releases, Figure 4. I-131 and Xe-133 equivalent source terms were used. These become somewhat conservative over the two-hour period due to decay of shorter-lived isotopes which was not considered. (The Xe-133 contribution dominates.) Again, for the large failed fuel fraction, the exposures may exceed guidelines and for the smaller failed fuel fraction, the contribution with degraded steam generator tube leakage is not insignificant.

TABLE 1.

AUG 29 1975

ATWS - Westinghouse

	t = 0 sec	13 sec	60	120	300	600	780	1020	1440.	4980	7250
	0 min	.2167 min	1	2	5	10	13	17	20	83	120
RCS											
T (°F)	514.65	610.	620	615	620	635	635	620	561	558	556
P (psia)	2250	2550	2500	2350	2350	2350.	2350	2350.	2350	2350	2350
h_f (BTU/lb)	591.	626.	641.2	635.	642.	666.	666	642	559.2	559.	559
v_f (ft³/lb)	.0235	.02345	.024	.024	.0241	.0251	.0251	.0241	.02156	.0215	.0215
ρ_f (lb/ft³)	42.62	42.64	41.67	41.67	41.49	39.84	39.84	41.49	46.38	46.5	46.5
Mass (lb)	537,500.	537,300.	525,500	525,500	522800.	502,500.	502,500	521,800.	584,415.	586,047.	586,047.
Power (Mwth)	3411.	3240.	2050.	1025.	650.	~~650.~~ 407	341.	229.	119.	68.	58.
Secondary											
T °F	533.3	565.6	570.	570	570	570	569	567	550.	550	550
P psia	910.	1185	1230	1230	1230	1230	1215	1200	1050	1050	1050
Water lb	88080.	89850.	89211.				89450.	89651.	91915.	91915.	91915.
Steam lb	2520	10140.	10589.				10426.	10265.	8815.	8815.	8815.
h_f (BTU/lb)	528.3	565.8	576.				574	571.7	550.1	550.1	550.1
h_g (BTU/lb)	1196.1	1186.5	1183.5				1184.2	1184.8	1191.	1191	1191.
v_f (ft³/lb)	.02127	.02227	.02243				.02237	.02232	.02177	.02177	.02177
v_g (ft³/lb)	.49490	.367	.3526				.357	.36245	.42224	.42224	.42224
ρ_f (lb/ft³)	47.0	45.0	44.6				44.7	44.8	45.9	45.9	45.9
ρ_g (lb/ft³)	2.02	2.72	2.84				2.80	2.76	2.37	2.37	2.37
Passenger release											
lb/min →		6250.	5899.	2400	667.	1200	4200	4200.	4200.	4200	4200.
lb →		1163.	3838.	2400.	2000.	6000.	~~[illegible]~~ 12600.	16,800.	54600.	211150.	55470.

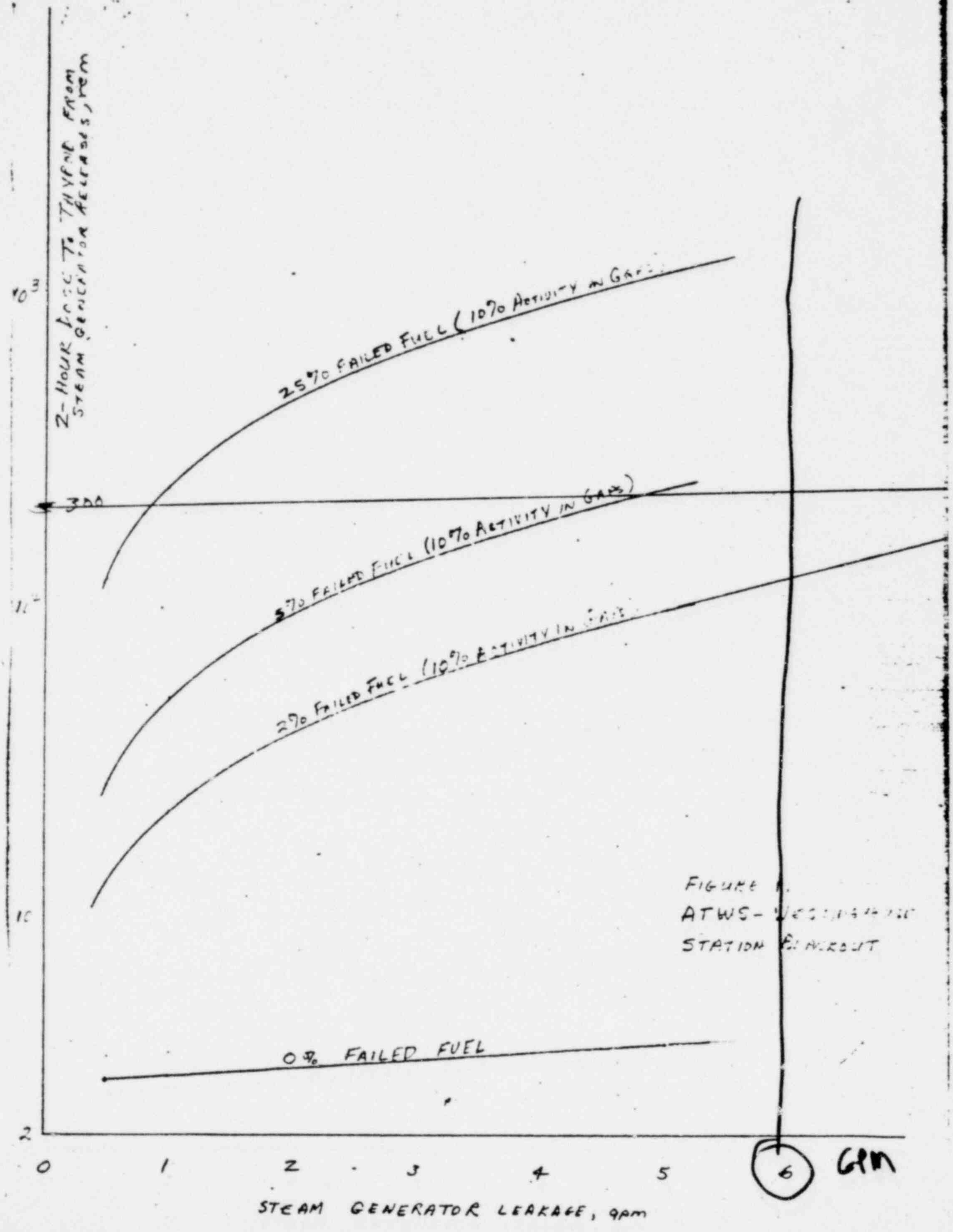

FIGURE 1.
ATWS-
STATION BLACKOUT

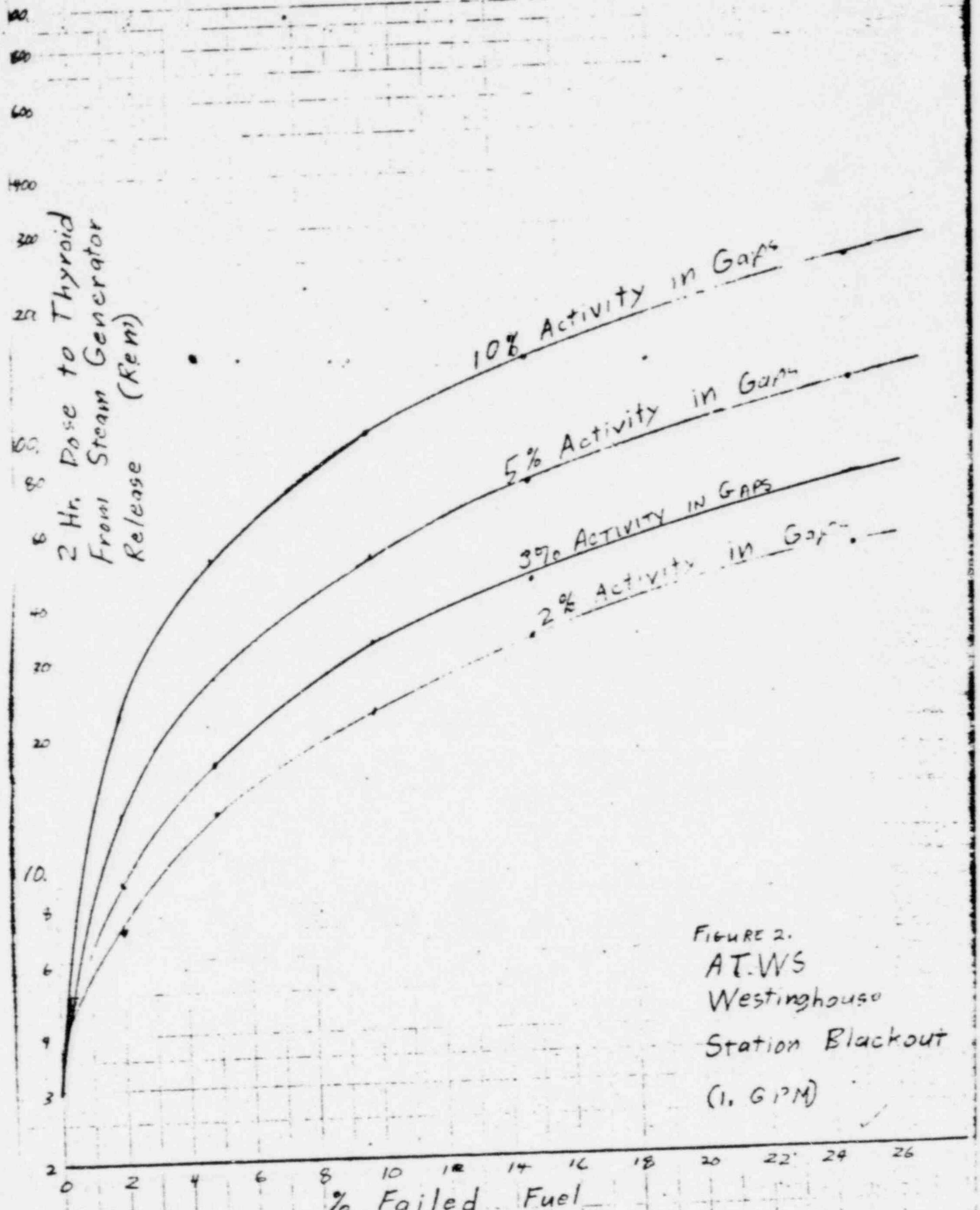

FIGURE 2.
ATWS
Westinghouse
Station Blackout
(1. GPM)

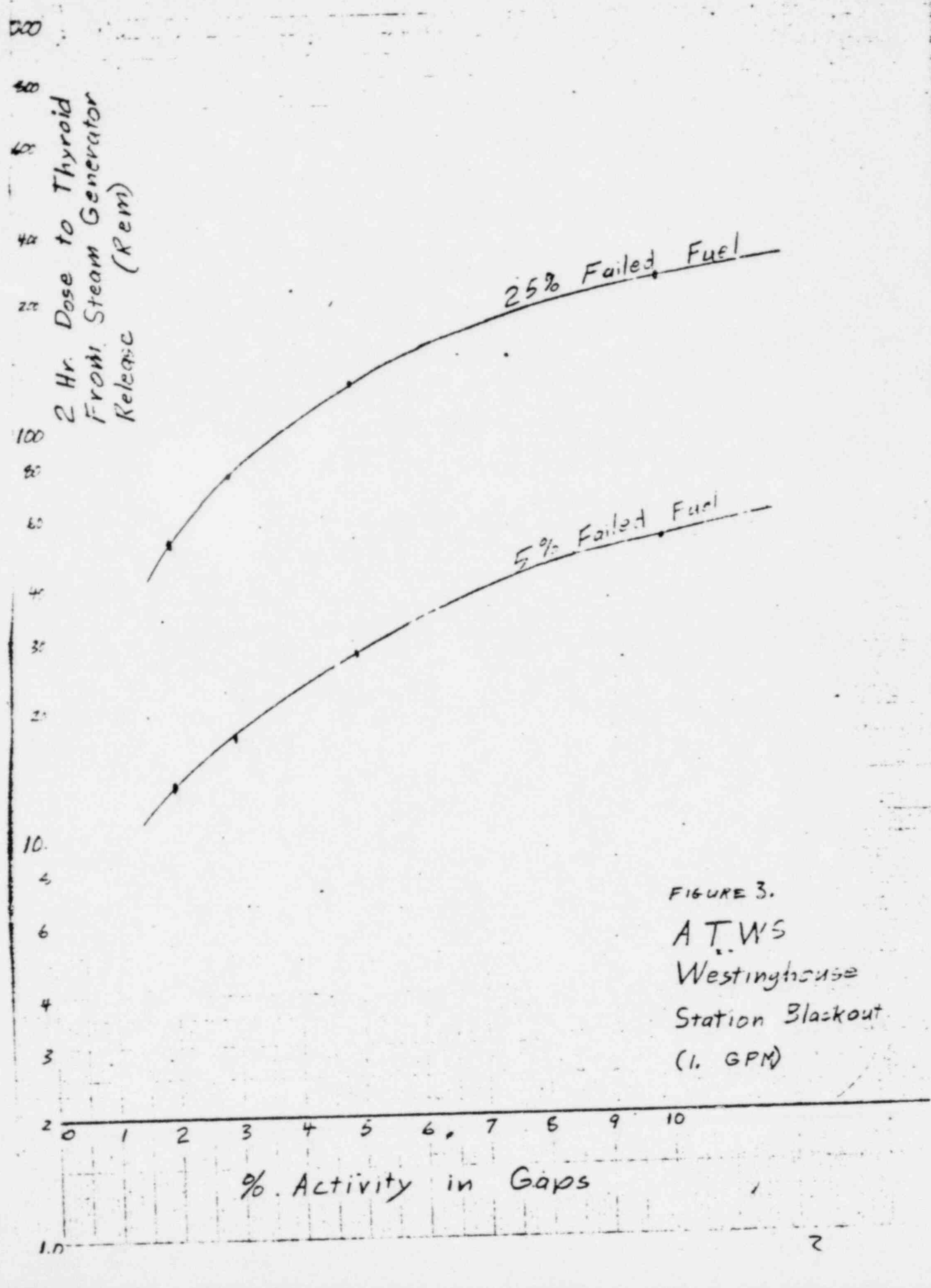

FIGURE 3.
ATWS
Westinghouse
Station Blackout
(1. GPM)

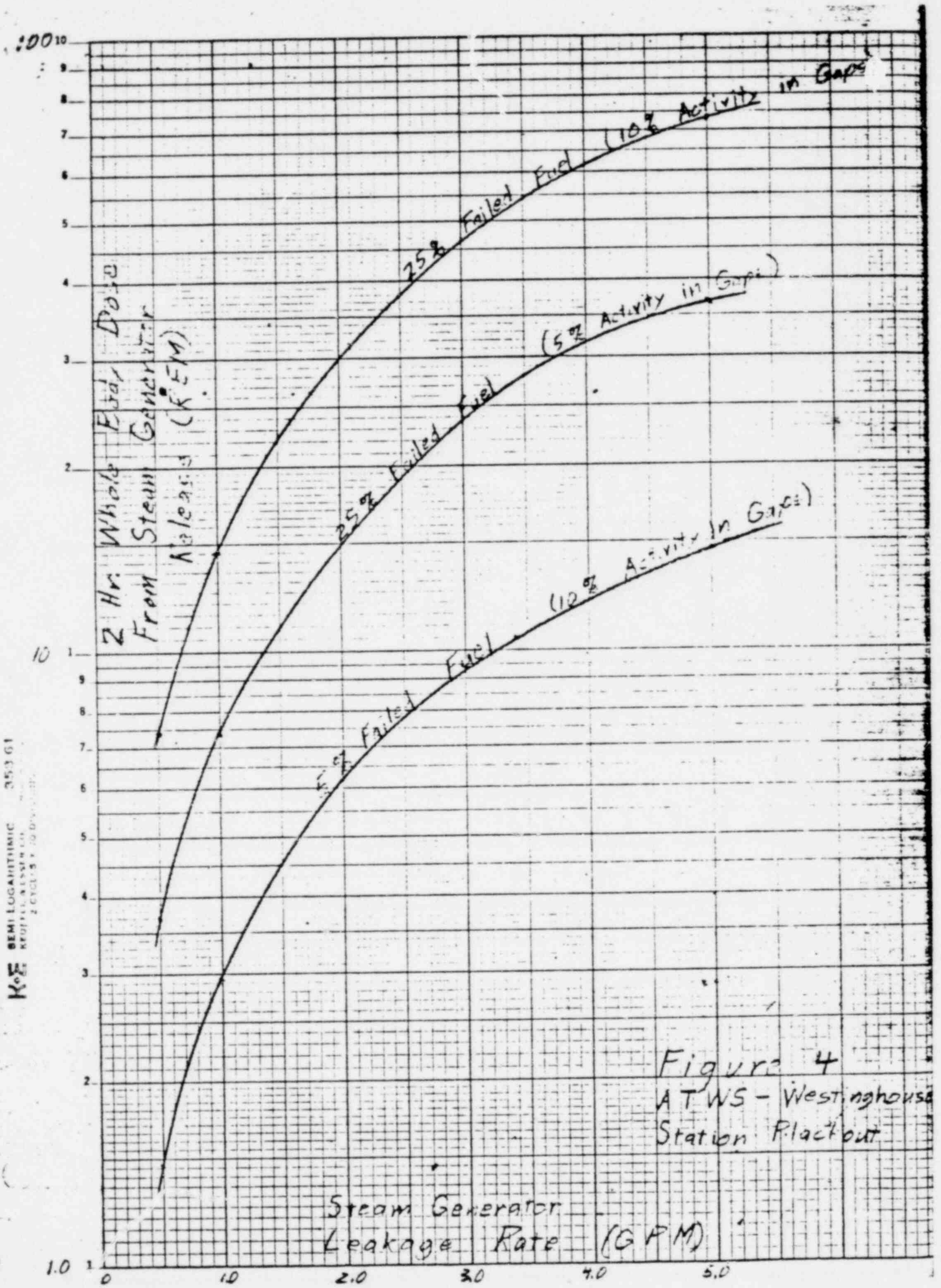

Figure 4
ATWS - Westinghouse
Station Blackout

TABLE 2

ESTIMATED CONSEQUENCES FROM CONTAINMENT LEAKAGE
FOLLOWING A LOSS OF NORMAL AC POWER
WITH FAILURE TO SCRAM

% Failed Fuel	% Activity in Gap	Doses, rem Thyroid	Doses, rem Whole Body
25	10	18	0.1
25	5	9	0.06
25	2	4	0.02
15	10	11	0.07
15	5	5	0.03
15	2	2	0.01
10	10	7	0.04
10	5	4	0.02
10	2	1	0.009
5	10	4	0.02
5	5	2	0.01
5	2	0.7	0.004
2	10	1	0.009
2	5	0.7	0.004
2	2	0.3	0.002

FILE S.1

Causes S-G tube failure in PWR.

it is desirable because of the international data basis. However, it is then very important that definitions and reporting practices are brought into close conformity if an international report is to serve a useful purpose.

Load factor and operation factor have been used in the IAEA report as basic performance factors. Unavailability factors, separate for planned and unplanned unavailability, are used as a measure of energy lost through a unit's outages and/or lower available capacity. When handling data it is important they be consistent and homogeneous. However, some ambiguity still remains in reporting the unavailability data, which may produce inconsistent results. It is recognized that there is difficulty in reporting unavailability in energy when it is related to a maximum capacity which may change several times during the year due to regulatory action, for example. In addition, there are different practices in reporting planned and unplanned unavailability among the Member States. The information contained in the report was made available to the Agency through designated national correspondents, and in some cases, directly from utilities. In the analytical reports, a classification is made by reactor type and size, i.e., HWR, GCR, PWR, and BWR, and below and above 600 MW(e). Performance, as measured by the weighted load factors as a function of the age of plants, has been calculated to determine whether they mature after a certain period to achieve design load factors. However, this kind of maturing process is not clearly evident except for the GCRs, while for all other types the increase in load factors with age does not appear to be significant. This may be due to the increased unavailability of older plants for major revisions, the generic problems appearing with some reactor types at specific times, and regulatory restrictions that affect the average value. One has to warn strictly here against drawing too far-reaching conclusions from too small a data base.

One problem experienced in the critical analysis of operating performance is the practical impossibility of determining the real duration of shutdown periods with reference to a particular outage category. Utility reports generally provide overall figures representing the total number of hours affecting a given shutdown, for which more than one outage category is indicated. This makes it impossible to identify the real shutdown duration, due to a referred outage category. The most common example is the refueling period, which may be extended to complete maintenance work which was anticipated or postponed according to management decisions.

An analysis of unavailabilities shows that maintenance and repair including refueling, although often amenable to planning, have caused a major part of the unavailability of all plant categories, and equipment failure has been a major cause of unplanned outages. The unavailability of large fossil-fueled power plants appears to be similar to nuclear power plants.

Lastly, a calculation of construction time span shows a considerable increase of construction time over the past year.[2]

1. "Operating Experience with Nuclear Power Stations in Member States," IAEA-127, 137, 150, 155, 168; STI/PUB/417 and STI/PUB/458, IAEA Annual Report.

2. "Performance Analysis of Nuclear Power Plants from Operating Experience Data," STI/PUB/452, IAEA Annual Report.

3. Condenser Tube Failures in Light-Water Reactors, *William L. Kershul (Nucl Power Exper, Inc)*

Agrees that simple

Fixes are all that needed. Same problems @ fossil plants

The Electric Power Research Institute has concluded that leakage resulting from condenser tube failures is a major contributor to nuclear plant reliability problems. Such leakage introduces contaminants into the steam cycle and results in corrosive and fouling effects on other components. A significant portion of the industry's steam generator problems is related to condenser tube failures.[1]

Condenser tube failures have been minimal at some nuclear power plants, while other plants have suffered extensive tube failures that have led to considerable amounts of downtime and, in some cases, to the enormous task of condenser retubing.

There are several condenser tube failure modes. One of the more prevalent means is excessive vibration, characterized by tube fretting, fatigue and failure of rolled joints. The vibration is usually caused by steam impingement on the tubes due to steam and water droplets which hit the tubes at high velocity. At Palisades in February 1972, 5 tubes failed due to resonant tube vibration. Each failure was a single, sharp, well-defined circumferential break at the midpoint between tube supports. Additional vibration necessitated the installation of >3000 stakes at the midpoint between tube supports. In addition, baffles were installed to preclude failure from steam impingement. At Brunswick 2 in April 1975, an improperly baffled steam line drain resulted in tube failures that required two shutdowns totaling ~65 h to effect repairs (~75 tubes were plugged). At Millstone 1, by August 1972, ~380 tubes had been plugged, of which many failed due to vibration. Similar failures occurred at Browns Ferry 1, Cooper, and other plants.

Plants using ocean water in the circulating water system have experienced saltwater corrosion and erosion. At Diablo Canyon 1 & 2, salt water left in the Cu-Ni tubes corroded them sufficiently allowing copper to be flushed out during pump tests, contributing to the death of 4000 to 14,000 abalone. The tubes were replaced with titanium in 1974. At Millstone 1 in 1972, the primary system became contaminated with excessive chlorides due to a large failure of the main condenser tubes. Eddy current testing of ~1450 Al-Brass tubes revealed 49% had undergone definite attack due to saltwater corrosion. Many pinholes had developed in the tubing. Retubing of the condenser with 70/30 Cu-Ni tubing was performed in late 1972 as this material demonstrated better performance. Some failures, along with inlet plugging, have been experienced since the retubing. Similar corrosion failures were experienced at Pilgrim.

Tube inlet erosion from high-velocity or turbulent seawater is another predominant cause of tube leakage. This problem was a common occurrence at San Onofre. Epoxy coating was applied to the inlet ends with only fair results. During the third refueling (Fall, 1973), 5754 tubes were replaced with 90/10 Cu/Ni. The original tubes were 90/10 and 70/30 Cu-Ni alloys. The inlet ends of the new tubes were allowed to protrude ~6 in. into the inlet plenum from the tubesheet. The theory in this installation was that sufficient streamlining would occur outside the tubesheet and mitigate to some extent eddying inside the tubesheet. Epoxy was applied to each inlet end to a depth of ~14 in. Acceptable wear resistance was obtained, but leakage was not completely arrested. In March 1975, 2 of the 4 condenser halves were retubed with titanium. Ten of these tubes failed and were plugged drift.

in November 1975. In addition, 9 suspected failed tubes were plugged. Similar erosion of the inlet ends of Al-Brass tubes was experienced at Pilgrim.

Broken LP turbine blades, vibrating impingement baffles, and other loose parts have caused tube failures. At San Onofre in 1969 and 1973, turbine blade failures caused condenser tube damage. Blade failures at Monticello and Maine Yankee caused similar tube failures. In 1972 at Maine Yankee, broken impingement baffles ruptured 3 condenser tubes and damaged 30 others. At Browns Ferry 1, a vibrating baffle wore holes in several condenser tubes which had to be replaced (20 tubes were also damaged at Peach Bottom 2 in this manner).

Poor condenser tube integrity and subsequent loss of power generation is undesirable. At Oyster Creek between 1970 and 1976, >50 load reductions and/or shutdowns were necessary for condenser tube repair. At Zion 1 & 2 (1974-75) there were 8 forced outages totaling ~750 h for condenser tube repair. It is evident that reliability and availability of the main condenser are of the utmost importance to maintaining good overall plant performance.

1. M. LEVENSON, *EPRI J.*, 2, 1 (1977).

Are excore detectors related and in θ direction? R. Sa

HDR

4. Operation of Fort Calhoun Station with a Reduced Complement of In-Core Detectors, *J. K. Gasper, R. L. Jaworski (Omaha Public Power Dist), R. R. Lee (C-E)*

Core performance monitoring at the Fort Calhoun Station utilizes 28 strings of self-powered nonmovable rhodium detectors with four detectors per string. In-core detector signals are processed using the GINCA[1] program. During the first 2000 MWd/MTU of Cycle 2 operation, signals were lost from more than 50% of the in-core detectors. The Technical Specifications related to the in-core system in effect at that time limited the reactor to 90% of full power. This paper describes an operation method and revisions of Technical Specification 2.10.3 implemented to allow full-power operation with up to 85% of the in-core detectors inoperable.

The in-core instrument system at Fort Calhoun Station is used in three ways to monitor core performance and to ensure operation within the limits used as initial conditions for the safety analysis:

1. to verify that the radial peaking factors do not exceed a specified value
2. to trigger alarms set on each individual instrument to ensure operation within specified peak linear heat rate limits
3. to determine the core internal axial shape index for periodic calibration of the ex-core detector system.

The revised Technical Specification allows full-power operation and ensures that safety margins and Limiting Conditions for Operation are not reduced when operating with a reduced complement of in-core detectors.

The new Specification requires a periodic comparison of instrument signals with core-follow calculations to ensure continued overall power distribution agreement between prediction and measurement if more than 25% of the in-core detectors are not operable. This comparison is made to verify that measured and calculated power distributions are within the uncertainty assumed in the safety analysis. If this comparison shows a higher uncertainty, then a penalty associated with the use of the in-core system for peak determination is applied to the appropriate setpoints to ensure that the confidence level in the localized peak power and DNBR margins is maintained.

The in-core detector alarm limits are normally set to ensure that the peak linear heat rate limit is not exceeded at each instrument location; however, if relatively large regions of the core are not monitored by in-core instruments, changes in the axial and radial distributions could occur without exceeding alarm limits. Therefore, if <75% of the in-core detectors are operable, each individual alarm limit is based on the same minimum margin to the limiting peak linear heat rate rather than the margin available at each instrument location. This, in effect, imposes a monitoring restraint on the power distribution as well as the peak linear heat rate and precludes the occurrence of peaks larger than the limit.

Loss of in-core detectors increases the core internal axial shape index uncertainty. This uncertainty is accounted for by applying a penalty to the peripheral axial shape index limits. If a specified minimum number of in-core detectors located in the interior and exterior regions of the core are not operable, the ex-core detectors are not calibrated and an additional penalty is applied to the peripheral axial shape index limits.

Cycle 2 operation concluded at full power using this Technical Specification. The uncertainty limit between measured and calculated in-core instrument signals was never exceeded in Cycle 2, even during the period when as few as 20% of the in-core detectors were operating.

This Technical Specification requires that a core follow program be maintained and that setpoint adjustment methodology be available in case the power distribution uncertainty is exceeded. The maintenance of these two programs is considered a small cost when compared to the loss of unit capacity.

Review

1. R. C. HELLENS, T. G. OBER, and R. D. OBER, "Method of Analyzing In-Core Detector Data in Power Reactors," *Trans. Am. Nucl. Soc.*, 12, 820 (1969).

5. On-Site Reconstitution of 108 St. Lucie I Fuel Assemblies, *K. N. Harris (Fla P&L-St. Lucie), J. J. Hutchinson (C-E), E. H. Smith, Jr. (C-E, Fla)*

PWR

The purpose of this paper is to describe in some detail the work that was accomplished in the spent fuel pool of Florida Power & Light's St. Lucie I Plant to replace 1300 poison shims in 108 fuel assemblies.

Chronological Summary of Events

1. Detected core power tilt @ 80% power—7/6/76.
2. Confirmed tilt and attendant growth at power. Reduced power for low-power physics testing—7/10/76.
3. Low-power physics tests confirmed a reactivity anomaly—commenced a plant shutdown/cooldown and disassembly—7/26/76.
4. Fuel and internals inspections began—8/1/76.
5. Fuel inspections completed—8/18/76.
6. Defueling completed—8/23/76.
7. Poison rod removal for analysis began 8/28/76.
8. Shipped 16 poison rods to ARMF/BMI—9/4/76.
9. Began fuel reconstitution—10/5/76.

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6 = MAX GPM

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FILE 5.1

11Ver King

$\chi/Q = 10^{-3}$

See also Prairie Island Memo

SEP 8 [illegible]

ATWS

V. Stello, Assistant Director for Reactor Safety

Prior to releases thru the pressurizer

WESTINGHOUSE ATWS CATEGORY B - LOSS OF NORMAL AC POWER

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~6 only

χ/Q

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Systems Judgement

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Conflict with typical, not DBA accident approach.

Brian K. Grimes, Chief
Accident Analysis Branch
Division of Technical Review
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See page 2

cc: R. Heineman
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E. Adensam
H. Fontecilla

$X/Q = 10^{-3}$

May be little difference between 1.0 & .01. DF.

WESTINGHOUSE ATWS
LOSS OF NORMAL AC POWER

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For which cases?

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MWt

1137 MWe

3

TABLE 1.

AUG 29 1975

ATWS - Westinghouse

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v_f (ft³/lb)	.0235	.02345	.024	.024	.0241	.0251	.0251	.0241	.02156	.0215	.0215
ρ_f (lb/ft³)	42.62	42.64	41.67	41.67	41.49	39.84	39.84	41.49	46.38	46.5	46.5
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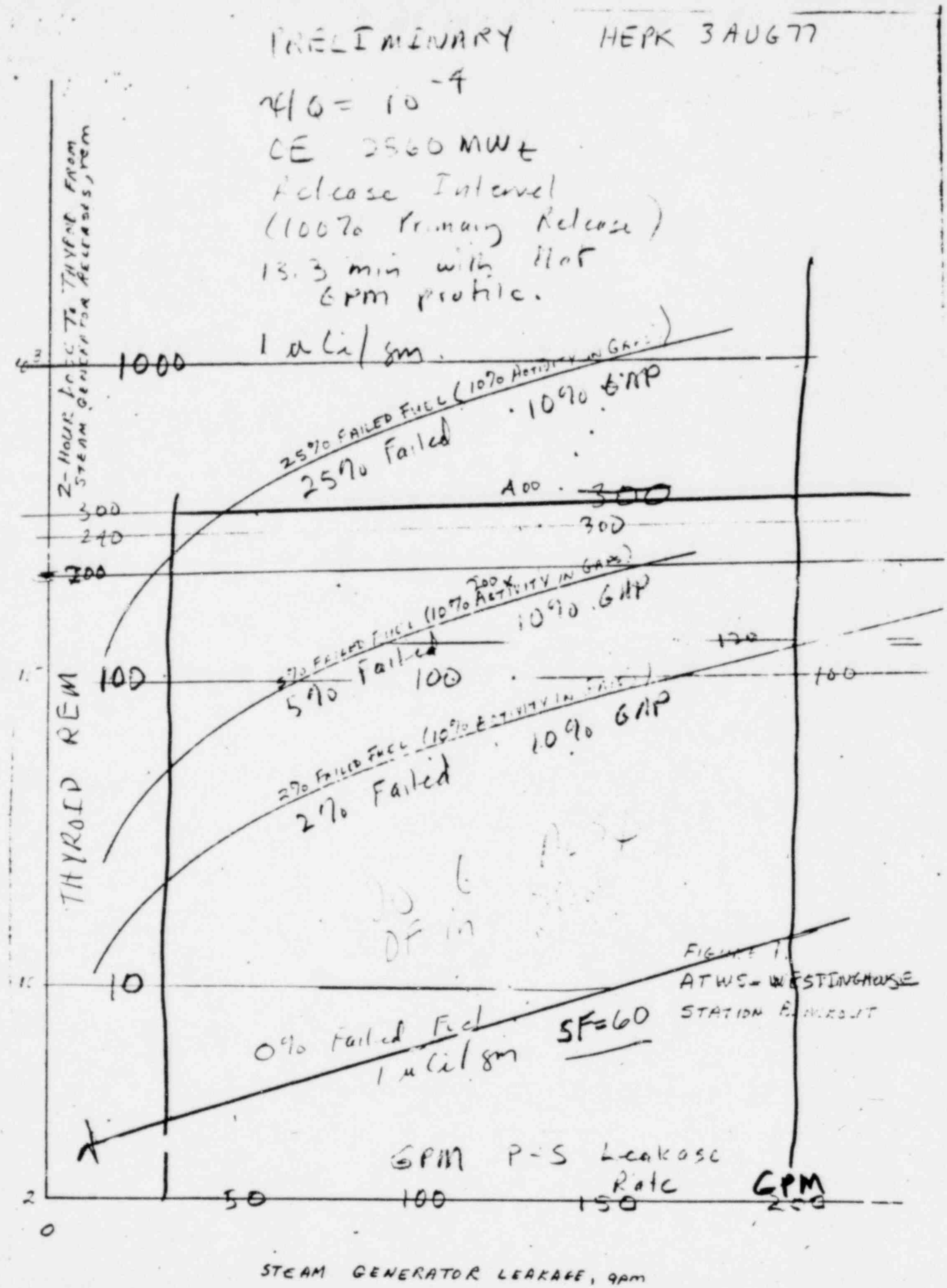
v_f (ft³/lb)	.02127	.02227	.02243				.02237	.02232	.02177	.02177	.02177
v_g (ft³/lb)	.49490	.367	.3526				.357	.36245	.42224	.42224	.42224
ρ_f (lb/ft³)	47.0	45.0	44.6				44.7	44.8	45.9	45.9	45.5
ρ_g (lb/ft³)	2.02	2.72	2.84				2.80	2.76	2.37	2.37	2.37
Pressurizer release											
lb/min →		6250.	5899.	2400	667.	1200	4200	4200.	4200.	4200	4200.
lb →		1463.	3838.	2400.	2000.	6000.	~~14,400.~~ 12,600.	16,800	54,600.	22250.	155470.


PRELIMINARY HEPK 3AUG77
X/Q = 10^-4
CE 2560 MWt
Release Interval
(100% Primary Release)
13.3 min with Hot
GPM profile.
1 uCi/gm.
2-HOUR DOSE TO THYROID FROM STEAM GENERATOR RELEASES, rem
THYROID REM
1000
300
240
200
100
10
25% FAILED FUEL (10% ACTIVITY IN GAP)
25% Failed
.10% GAP
400
300
5% FAILED FUEL (10% ACTIVITY IN GAP)
5% Failed
10% GAP
2% FAILED FUEL (10% ACTIVITY IN GAP)
2% Failed
10% GAP
120
100
0% Failed Fuel
1 uCi/gm
SF=60
FIGURE 1.
ATWS - WESTINGHOUSE
STATION BLACKOUT
GPM P-S Leakage Rate
GPM
0
50
100
150
200
STEAM GENERATOR LEAKAGE, gpm

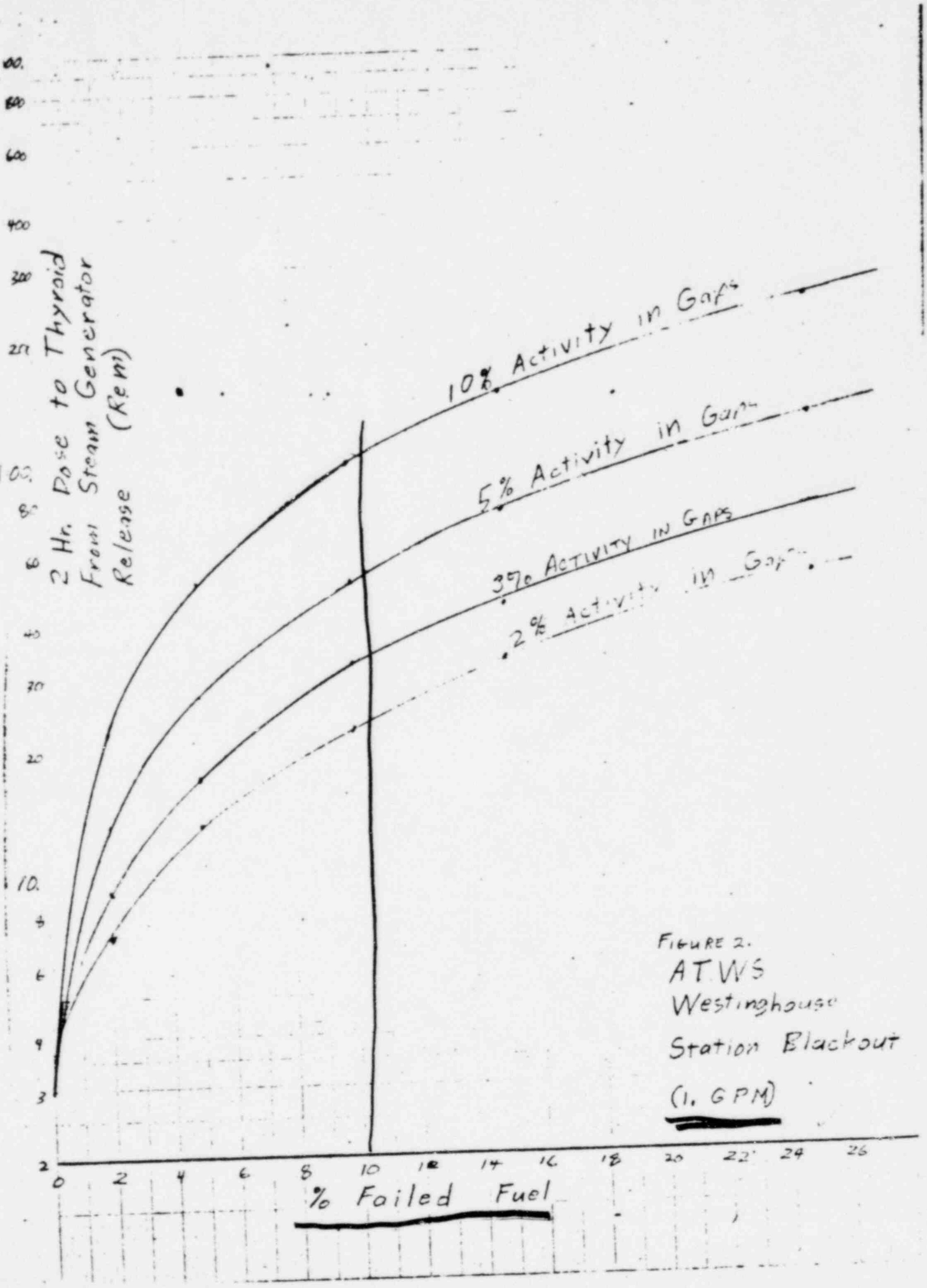


FIGURE 2.
ATWS
Westinghouse
Station Blackout
(1. GPM)

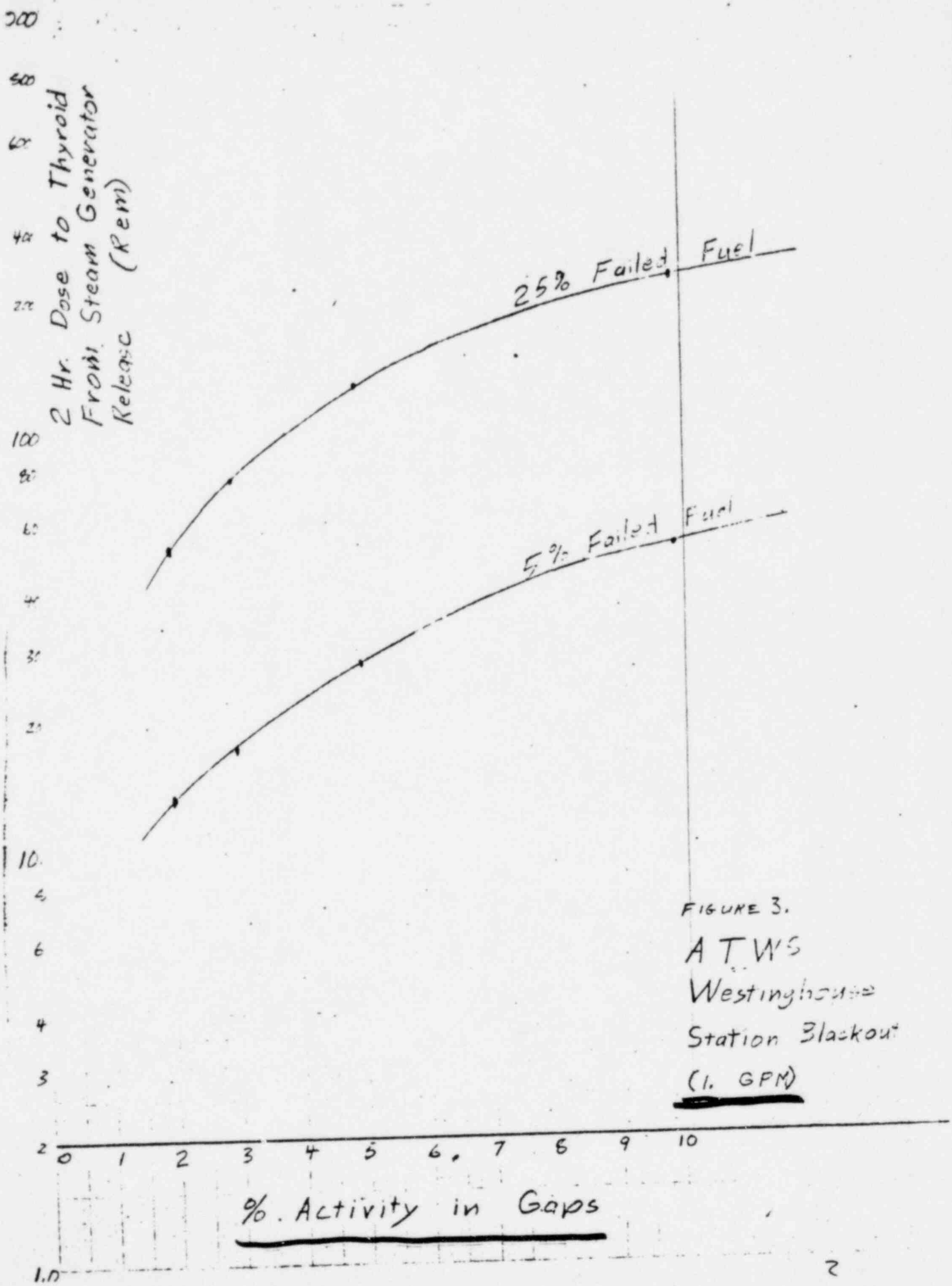


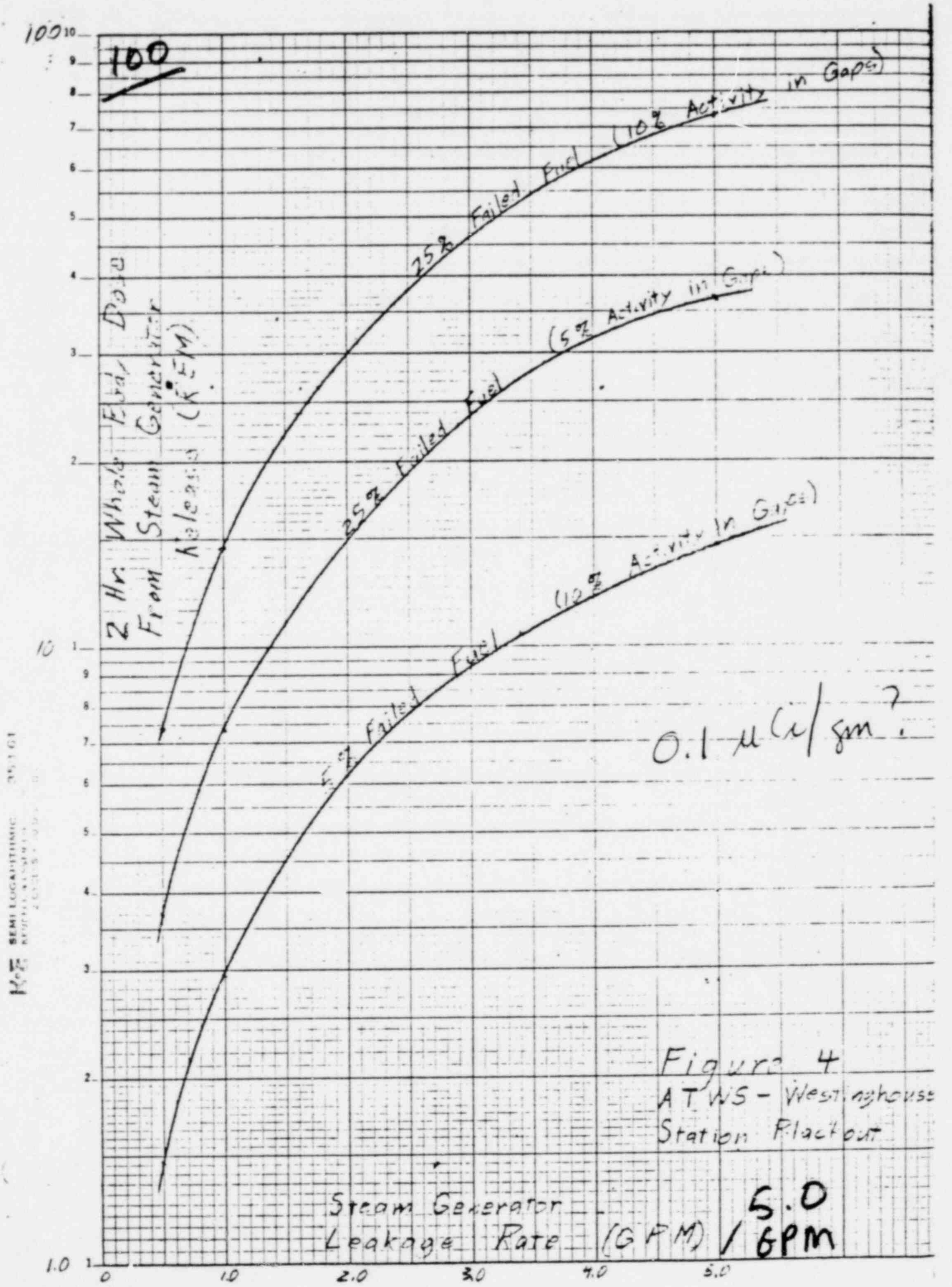
FIGURE 3.

ATWS

Westinghouse

Station Blackout

(1. GPM)


100
2 Hr. Whole Body Dose From Steam Generator Releases (REM)
25% Failed Fuel (10% Activity in Gaps)
25% Failed Fuel (5% Activity in Gaps)
5% Failed Fuel (10% Activity in Gaps)
0.1 μCi/gm?
Figure 4
ATWS - Westinghouse
Station Blackout
Steam Generator Leakage Rate (GPM)
5.0 / GPM
100
10
1.0
0
1.0
2.0
3.0
4.0
5.0

Containment Leakage

TABLE 2

ESTIMATED CONSEQUENCES FROM CONTAINMENT LEAKAGE
FOLLOWING A LOSS OF NORMAL AC POWER
WITH FAILURE TO SCRAM

% Failed Fuel	% Activity in Gap	Doses, rem Thyroid	Doses, rem Whole Body
25	10	18	0.1
25	5	9	0.06
25	2	4	0.02
15	10	11	0.07
15	5	5	0.03
15	2	2	0.01
10	10	7	0.04
10	5	4	0.02
10	2	1	0.009
5	10	4	0.02
5	5	2	0.01
5	2	0.7	0.004
2	10	1	0.009
2	5	0.7	0.004
2	2	0.3	0.002