August 9, 1984

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY	)	Docket No. 50-400 OL
and NORTH CAROLINA EASTERN	)	
MUNICIPAL POWER AGENCY	)	
	)	
(Shearon Harris Nuclear Power Plant)	)	

APPLICANTS' TESTIMONY OF MICHAEL J. HITCHLER IN RESPONSE TO JOINT INTERVENORS CONTENTION VII (4) (STEAM GENERATOR TUBE RUPTURE ANALYSIS)

Q.1 Please state your name, address, present occupation and employer.

A.1 My name is MICHAEL JOHN HITCHLER. I am Manager of Plant Risk Analysis with the Nuclear Safety Department of Westinghouse Electric Corporation, P. O. Box 355, Pittsburgh, Pennsylvania 15230.

Q.2 State your educational background and professional work experience.

A.2 I was graduated from Lowell Technological Institute in 1974 with a Bachelor of Science Degree in Nuclear and Mechanical Engineering and from Carnegie-Mellon University in 1978 with a Master of Science Degree in Mechanical Engineering.

I have published five articles in various technical periodicals and have authored or coauthored eight Westinghouse reports which pertained to reactor accident analyses, emergency/abnormal operating instruction development and probabilistic risk analyses.

I joined Westinghouse in June 1975 as an Engineer. I was promoted to Senior Engineer in December 1978. My responsibilities during that time included performing accident analyses for use in licensing documents. I have served as a Westinghouse liaison with the NRC, architect engineers and utilities for issues concerning reactor protection system design requirements. My specific areas of specialization included core and systems response to transients initiated in the primary system, development of methodology for safety analysis of reload cores, and simulation of actual plant transients for computer verification purposes. I also had the lead responsibility for the transfer of the above technology to various utility customers. This responsibility included the structuring of classroom as well as on-the-job training for a number of utility personnel.

In June 1981, I was assigned responsibilities in the risk assessment area. These responsibilities involved the development and implementation of strategic programs to enhance and to apply risk assessment technology for use in nuclear power plant design and licensing. This work included development and quantification of event trees for use by the Westinghouse Owner's Group in reviewing emergency and abnormal operating procedures as part of its response to post TMI issues. I assisted in the development and review of Auxiliary Feedwater System Reliability Studies for three nuclear plants.

In October 1981, I was promoted to the position of Manager, Probabilistic Risk Assessment (PRA) Group. I presently have lead responsibility for a probabilistic risk study of two non-domestic, pre-construction nuclear stations, which includes development of a risk baseline and an assessment of potential design alternatives. I have also worked on three domestic station risk studies, contributing extensively in the following areas: plant and containment event tree construction, systems success criteria for fault tree development, external (seismic, wind, fire, etc.) event analysis and review of the results sections.

I am a member of the American Nuclear Society (ANS) and the American Society of Mechanical Engineers. I served on two ANS Standards committees and contributed to several Atomic Industrial Forum (AIF) and Institute of Electrical and Electronics Engineers (IEEE) committees on development of risk criteria and utilization of PRA approach to licensing.

Q.3 Please elaborate on your professional experience that is directly relevant to the testimony which you are presenting regarding steam generator tube rupture events.

A.3 I have been involved in developing probabilistic models to quantify the frequency of steam generator tube rupture events, and their consequences in terms of core melt frequency and public risk, since 1982. I have directed the performance of PRA analyses of tube rupture events for the Byron, Millstone 3, Sizewell B (British), and PUN (Italian) nuclear power stations.

Q.4 What is the purpose of your testimony?

A.4 The purpose of my testimony is to address the one remaining issue in this proceeding raised by Joint Intervenors Contention VII -- i.e., the allegation that Applicants' steam generator tube rupture analysis found in the Final Safety Analysis Report is inadequate because it fails to consider multiple tube rupture events.

Q.5 Describe the steam generator tube rupture event that is analyzed by Applicants in the Harris Plant Final Safety Analysis Report (FSAR).

A.5 The Harris FSAR contains an analysis of a single double-ended rupture of a steam generator tube, consistent with Section 15.6.3 of the NRC "Standard Review Plan," NUREG-0800, Revision 3.

Q.6 Steam generator tube rupture events are defined as "Condition IV" events in Section 15.0 of the Harris FSAR. What is a Condition IV event?

A.6 "Condition IV" events are defined as faults which are not expected to take place during the lifetime of the plant. In other words, the frequency of these events is judged to be less than once in 40 years, or less than 2.5×10^{-2} per year.

Q.7 Is this characterization of a steam generator tube rupture as a Condition IV event consistent with the operating history of Westinghouse pressurized water reactors (PWR)?

A.7 This characterization is consistent with PWR performance in the approximately 233 plant years of experience to date. As I will explain below, based on historical experience alone, the frequency of steam generator tube ruptures is predicted to be no more than once in 45 years of operation.

Q.8 What is the total number of tube years of experience in Westinghouse-design nuclear plants with Inconel steam generator tubes similar to the tubes in the Harris Plant steam generators?

A.8 The total number of tube years of experience in Westinghouse-design plants with Inconel steam generator tubes was determined, based on data through July 1983, as shown in Tables 1 through 6.

These tables cover different categories of plants and set forth plant designation, number of tubes, date of commercial operation, and total calendar years between beginning of commercial operation and July 1983. The data in these tables show a total of over four million tube years of experience since the beginning of commercial operation. For purposes of our analysis here, these data were discounted 10 percent to 3.6 million tube years.

Q.9 How many tube rupture events have actually occurred in Westinghouse-design nuclear plant steam generators?

A.9 Table 7 presents a list of tube rupture events that have occurred in Westinghouse steam generators. All five of these events had flow rates large enough to cause plant trip and initiate safety injection. Only one event, however, had a flow rate that even approximates a full double-ended tube rupture as described in the FSAR; the other four events were much smaller in magnitude.

Q.10 Based on this historical data alone, what would be the predicted failure rate for steam generator tubes in Westinghouse type PWRs?

A.10 With five tube ruptures in an experience base of 3.6×10^6 tube years, the experienced tube rupture failure rate would be $\lambda = 5 \div (3.6 \times 10^6) = 1.4 \times 10^{-6}$/tube-year or, using Chi-square tables, the 50 percent confidence value would be

$$\lambda_{50 \text{ percent}} = \frac{11.34}{2 \times 3.6 \times 10^6} = 1.6 \times 10^{-6}\text{/tube-year}$$

with upper and lower 95 percent confidence limits of

$$\frac{21.03}{2 \times 3.6 \times 10^6} \geq \lambda \geq \frac{5.23}{2 \times 3.6 \times 10^6}$$

$$2.9 \times 10^{-6} \geq \lambda \geq 0.73 \times 10^{-6} \text{ per tube-year}$$

Based on this calculation, the tube failure rate derived from experience is 1.6×10^{-6}/tube-year. This is equivalent to the figure of one failure in 45 years that I mentioned previously. It could be as low as 0.73×10^{-6} or as high as 2.9×10^{-6} per tube-year.

Q.11 Is there any reason to believe that the steam generator tube failure rate for the Harris Plant steam generators is likely to be better than the historical average?

A.11 Yes, because of advances in the state of the art in the design, operation, and inspection of steam generators, it is believed that nuclear plants utilizing Model D steam generators, such as Shearon Harris will be less likely to experience steam generator tube failure. Cogent reasons can be given as to why certain of the five tube ruptures experienced to date should not occur in the Model D steam generators since the operating conditions at certain of the plants which have experienced tube ruptures are not applicable to the Harris Plant. Cogent reasons can also be given as to why the occurrence rate should be substantially less because of design and inspection advancements. These are described below.

Q.12 What were the causes of the five steam generator tube rupture events experiened in Westinghouse-design plants?

A.12 At Plant E[1] in February 1975, phosphate wastage had thinned tubes in a zone just above the tubesheet where sludge had collected. In addition to thinning, some stress corrosion cracking was also present. The events at Plant I in September 1976, and Plant bb in June 1979, show some similarities.

In both cases, the tubes had suffered stress corrosion cracking starting from the primary side. At Plant I, this was due to denting accompanied by "hour glassing" of the flow slots. At Plant bb, the affected tube had excessive ovality which led to high stresses at the U-bend. The two remaining events, at Plant N in October 1979, and Plant C in January 1982, were both due to foreign objects fretting and wearing the tube along one side.

Q.13 Why do you believe that the changes which have been incorporated into the design and operation of Harris Plant steam generators are likely to reduce the steam generator tube failure rate?

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1 Plant designations refer to notation used in Tables 1 through 6.

A.13 Due to advances in the design of Model D steam generators and in operations, maintenance, and inspection procedures at Harris, tube failure resulting from these causes is judged to be reduced in frequency[2]. The phosphate wastage, for example, has been eliminated since phosphates will not be used at Harris, thus the tube rupture frequency attributed to wastage is judged to be lowered by at least a factor of 100. A reduction factor is utilized even though phosphate wastage is impossible at Harris, because other types of chemical wastage (currently unobserved) may still be possible.

Denting of tubes, if it occurs at all, will develop much more slowly and to a more limited extent than in steam generators at other plants because of:

- plant operation with only AVT chemistry control;
- reduction of copper in the secondary side systems as compared to other plants;
- fresh water condenser cooling with resultant decrease in chloride concentrations as compared to plants operating on sea or brackish water.

Stress corrosion cracking (SCC) at Harris is judged very unlikely because of the following:

- limitation of the use of copper which decreases the rate of SCC by reducing the concentration of alkaline salts; and

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2 (Note of Counsel) These design advances and operational commitments were described in detail in the affidavits of Thomas E. Timmons, Glenn E. Lang, and Alan B. Cutter, filed in support of "Applicants' Motion for Partial Summary Disposition of Joint Contention VII (Steam Generators)." The Board granted Applicants' motion and the factual issues addressed therein are not in dispute. <u>See</u> Tr. 2167-68 (Conference Call, July 12, 1984).

- design advances which (a) minimize crevices between the tube and tubesheet through full depth expansion of tubes and (b) provide features to reduce the accumulation of overlying sludge.

In addition, any tube degradation at Harris will most likely be identified before rupture could occur due to extensive In-Service Inspection which includes: full inspection of all tubes before the plant is put into operation, eddy current testing, ultrasonic inspection techniques, profilometry probes, and continuous monitoring of water quality, radioactivity, leakage rates, etc. For these reasons, tube rupture due to denting and SCC is judged to be reduced by a factor of five.

One type of tube leakage event which is not affected by design advances is wear due to foreign objects, which was responsible for the two largest tube rupture events which have occurred. However, due to rigorous quality assurance procedures as well as monitoring for loose parts at Harris, this type of tube leakage event is judged to be much less likely than historical frequency indicates, and a lowering by a factor of two is assumed in this study.

Implementation of the modifications to minimize tube vibration in the Model D-4 steam generators should reduce tube vibration levels such that they will be at or below the levels contained in the experience base used in this analysis.

Q.14 Based on the improvements incorporated into the Harris Plant steam generator design and operation, what steam generator tube failure rate would you predict for the operation of Harris steam generators?

A.14 Given the design, maintenance inspection technique and operating advances described above, the number of historical tube rupture incidents which are applicable to Harris for this analysis can be decreased from five to about 1.5 (virtually none due to phosphate wastage, 0.4 due to denting and SCC, and one due to loose parts).

Table 8 shows how the 50 and 95 percent confidence level failure rate decreases as the number of tube ruptures in the experience base to the present decreases.

On this basis, the median (50 percent confidence level) failure rate would be $\lambda_{50\ percent} = 0.6 \times 10^{-6}$/tube-year. Although the above approach utilizes some engineering judgment in conjunction with the experience base, the data available and identified advances provide reasonable support for this. In fact, engineering judgment would suggest that the advances in the state of the art should yield an even lower failure rate.

This failure rate of 0.6×10^{-6}/tube-year corresponds to an annual frequency of 8.2×10^{-3} per year

$$\left(\frac{0.6 \times 10^{-6}}{\text{tube-year}} \times \frac{4578 \text{ tubes}}{\text{SG}} \times 3 \text{ SG} = \frac{8.2 \times 10^{-3}}{\text{year}}\right)$$

at Harris, or one event in approximately 120 years of reactor operation. This predicted value is significantly below the historical base. Thus the operation of Model D-4 steam generators at Harris as compared with previous experience should result in an even higher degree of public safety with respect to these issues.

Q.15 Why shouldn't multiple tube rupture events be considered in analyses of design basis accidents?

A.15 Multiple tube rupture events should not be considered in analyses of design basis accidents due to their low frequency of occurrence and due to their insignificant contribution to risk.

Q.16 Have you determined the frequency of multiple tube ruptures in Westinghouse PWRs?

A.16 Analyses have been performed to assess the frequency of multiple tube ruptures in Westinghouse PWRs. Since a multiple tube rupture has never occurred, a probabilistic model based on pressure differentials across the steam generator tubes was developed to evaluate the frequency of these events.

Q.17 Briefly describe the "pressure pulse" model developed to evaluate the frequency of multiple tube ruptures.

A.17 The "pressure pulse" model relates the pressure differential across steam generator tubes to tube failure probability. Based on laboratory testing, the minimum tube burst capability at the beginning of tube life is assessed at 10,000 psi. The tubes are assumed to degrade linearly from 0 to 40 years of service life.

The model applies a conservative distribution to the individual tube failure probability; the binomial distribution is then used to calculate the probability that one, two, or three tubes fail. The model assumes that during normal reactor operation, transient pressure swings up to about the 2500 psia safety valve set point occur with a frequency of once per year. The "pressure pulse" model is described in detail in Exhibit A.

This model was used to estimate the frequency of single and multiple tube ruptures. The calculated single tube rupture frequency of 7.5×10^{-3} per year is consistent with the value of 8.2×10^{-3} per year calculated from tube experience data.

Q.18 What do you calculate the multiple tube rupture frequency to be for steam generators at the Harris Plant?

A.18 Using the "pressure pulse" model described above, the multiple tube rupture frequency calculated for the Harris Plant is 7×10^{-5} per year. This corresponds to one such event in about 14,000 plant years.

Q.19 Does the risk of multiple tube rupture events contribute significantly to overall risk for the Harris Plant?

A.19 A number of PRA studies have been performed in the United States and Europe which have evaluated the risk to the public from single and multiple steam generator tube rupture initiating events. Results of these PRA studies show that tube ruptures would not contribute significantly to overall risk for a plant such as Shearon Harris.

Based on results of PRA analyses, the Harris core melt frequency due to tube rupture initiating events was estimated to be about 3×10^{-7} per year. Of this frequency, three percent (1×10^{-8} per year) is due to multiple tube rupture events. Applying representative PRA consequence models, the public risk from multiple tube rupture events is judged to be an insignificant contributor to overall plant risk at a plant such as Shearon Harris.

Q.20 Is this assessment of the low risk of tube rupture events consistent with independent evaluations of the NRC?

A.20 This assessment is consistent with the independent NRC evaluation performed in draft NUREG-0844, which concludes that SGTR events beyond the design basis do not contribute a significant fraction of the risks associated with other reactor events at a given site.

Q.21 What are your conclusions regarding the frequency of multiple tube ruptures at the Harris Plant?

A.21 Based on the analysis described above and my experience in other assessments, I am confident that multiple tube rupture events will not contribute significantly to overall public risk at Harris. Due to the relatively insignificant contribution of multiple tube ruptures to public risk, there is little benefit to be gained from performing a vigorous analysis of the consequences of such an event. This assessment reflects the significant design improvements that have been incorporated in Westinghouse Model D-4 steam generator and the improvements in steam generator operations, maintenance and inspections which provide additional assurance of the safe operation of the Harris Plant.

TABLE 1

STEAM GENERATOR TUBE EXPERIENCE TO JULY 1983
U.S. WESTINGHOUSE INCONEL PLANTS

Plant	No. of Tubes	Commercial Operation	Years	Tube-Year
A	11,382	1/68	15.4	$17.5 \times 10^{+4}$
B	15,176	1/68	15.4	$12.4 \times 10^{+4}$
C	6,520	3/70	13.2	$8.6 \times 10^{+4}$
D	9,780	3/71	12.2	$11.9 \times 10^{+4}$
E	6,520	12/70	12.5	$8.2 \times 10^{+4}$
F	6,520	10/72	10.7	$7.0 \times 10^{+4}$
G	10,164	12/72	10.5	$10.7 \times 10^{+4}$
H	9,780	12/73	9.5	$9.3 \times 10^{+4}$
I	10,164	5/73	10.1	$10.3 \times 10^{+4}$
J	13,040	7/74	8.9	$11.6 \times 10^{+4}$
K	13,552	10/73	9.7	$13.1 \times 10^{+4}$
L	9,780	9/73	9.7	$9.5 \times 10^{+4}$
M	13,552	9/74	8.7	$11.8 \times 10^{+4}$
N	6,776	12/73	9.5	$6.4 \times 10^{+4}$
O	6,776	6/74	9.1	$6.2 \times 10^{+4}$
P	6,776	12/74	8.5	$5.8 \times 10^{+4}$
Q	13,552	8/75	7.8	$10.6 \times 10^{+4}$
R	13,552	5/76	7.1	$9.6 \times 10^{+4}$
S	13,040	8/76	6.8	$8.9 \times 10^{+4}$
T	10,164	4/77	6.2	$6.3 \times 10^{+4}$
U	13,552	6/77	6.0	$8.1 \times 10^{+4}$
V	10,164	12/77	5.5	$5.6 \times 10^{+4}$
W	13,552	7/78	4.9	$6.6 \times 10^{+4}$
X	10,164	6/78	5.0	$5.1 \times 10^{+4}$
Y	10,164	12/80	2.5	$2.5 \times 10^{+4}$
Z	13,552	7/81	1.9	$2.6 \times 10^{+4}$
A1	13,552	10/81	1.7	$2.3 \times 10^{+4}$

TABLE 1 (Continued)

STEAM GENERATOR TUBE EXPERIENCE TO JULY 1983

U.S. WESTINGHOUSE INCONEL PLANTS

Plant	No. of Tubes	Commercial Operation	Years	Tube-Year
A2	10,164	7/81	1.9	$1.9 \times 10^{+4}$
A3	18,696	12/81	1.5	$2.8 \times 10^{+4}$
A4	13,552	6/82	1.0	$1.4 \times 10^{+4}$
Total			233.4	245.6×10^{4} Tube Years

TABLE 2

STEAM GENERATOR TUBE EXPERIENCE TO JULY 1983
WESTINGHOUSE FOREIGN PLANTS (INCONEL)

Plant	No. of Tubes	Commercial Operation	Years	Tube-Year
AA	2,604	8/69	13.8	$3.6 \times 10^{+4}$
BB	5,208	12/69	13.5	$7.0 \times 10^{+4}$
CC	5,208	3/72	11.2	$5.8 \times 10^{+4}$
DD	10,164	11/74	8.6	$8.7 \times 10^{+4}$
EE	10,164	5/75	8.1	$8.2 \times 10^{+4}$
FF	6,776	4/78	5.2	$3.5 \times 10^{+4}$
GG	13,552	3/79	4.2	$5.7 \times 10^{+4}$
HH	14,022	4/81	2.2	$3.1 \times 10^{+4}$
II	14,022	12/81	1.5	$2.1 \times 10^{+4}$
JJ	9,156	12/81	1.5	$1.4 \times 10^{+4}$
	Total		69.8	49.1×10^{4} Tube Years

TABLE 3

STEAM GENERATOR TUBE EXPERIENCE TO JULY 1983
MHI PLANTS

Plant	No. of Tubes	Commercial Operation	Years	Tube-Year
ZZ	6,520	7/72	10.9	$7.1 \times 10^{+4}$
YY	10,164	11/75	7.6	$7.7 \times 10^{+4}$
XX	6,776	10/75	7.7	$5.2 \times 10^{+4}$
WW	10,164	12/76	6.5	$6.6 \times 10^{+4}$
VV	6,776	9/77	5.7	$3.9 \times 10^{+4}$
UU	6,776	3/81	2.2	$1.5 \times 10^{+4}$
TT	6,776	3/82	1.2	$0.8 \times 10^{+4}$
Total			41.8	32.8×10^{4} Tube Years

TABLE 4

STEAM GENERATOR TUBE EXPERIENCE TO JULY 1983
FRAMATOME PLANTS

Plant	No. of Tubes	Commercial Operation	Years	Tube-Year
a	10,164	12/77	5.5	$5.6 \times 10^{+4}$
b	10,164	3/78	5.2	$5.3 \times 10^{+4}$
c	10,164	2/79	4.3	$4.4 \times 10^{+4}$
d	10,164	2/79	4.3	$4.4 \times 10^{+4}$
e	10,164	7/79	3.9	$4.0 \times 10^{+4}$
f	10,164	12/79	3.5	$3.6 \times 10^{+4}$
g	10,164	11/80	2.6	$2.6 \times 10^{+4}$
h	10,164	12/80	2.5	$2.5 \times 10^{+4}$
i	10,164	9/80	2.7	$2.7 \times 10^{+4}$
j	10,164	12/80	2.5	$2.5 \times 10^{+4}$
k	10,164	12/80	2.5	$2.5 \times 10^{+4}$
l	10,164	6/81	2.0	$2.0 \times 10^{+4}$
m	10,164	5/81	2.1	$2.1 \times 10^{+4}$
n	10,164	2/81	2.3	$2.3 \times 10^{+4}$
o	10,164	5/81	2.1	$2.1 \times 10^{+4}$
p	10,164	12/82	1.5	$1.5 \times 10^{+4}$
q	10,164	10/81	1.7	$1.7 \times 10^{+4}$
r	10,164	11/81	1.6	$1.6 \times 10^{+4}$
s	10,164	12/81	1.5	$1.5 \times 10^{+4}$
t	10,164	11/82	0.6	$0.6 \times 10^{+4}$
Total			54.9	55.5×10^{4} Tube Years

TABLE 5

STEAM GENERATOR TUBE EXPERIENCE TO JULY 1983
MISCELLANEOUS WESTINGHOUSE LICENSEE PLANTS

ACECOWEN

Plant	No. of Tubes	Commercial Operation	Years	Tube-Year
aa	6,520	2/75	8.2	$5.3 \times 10^{+4}$
bb	6,520	11/75	7.6	$5.0 \times 10^{+4}$

ACLF

Plant	No. of Tubes	Commercial Operation	Years	Tube-Year
cc	10,164	9/75	7.7	$7.8 \times 10^{+4}$
Total			23.5	18.1×10^{4} Tube Years

TABLE 6

SUMMARY OF STEAM GENERATOR TUBE EXPERIENCE TO JULY 1983

	No. of Plants	Plant-Years	Tube-Years
Westinghouse (Inconel Tube)			
US plants	31	233.4	2,456,000
Foreign plants	10	69.8	491,000
Subtotal	41	303.2	2,947,000
Westinghouse Licensee plants			
MHI	7	41.8	328,000
FRA	20	54.9	555,000
Miscellaneous W Licensee Plants	3	23.5	181,000
Subtotal	30	120.2	1,064,000
TOTAL	71	423.4	4,011,000

TABLE 7

TUBE RUPTURE EXPERIENCES SUMMARY

No.	Occurrence Date	Plant	Attributed Cause	Estimated Leak Rate
1	Feb. 26, 1975	E	Phosphate Wastage + SCC	125 gpm (1)
2	Sept. 15, 1976	I	Denting + SCC	80 gpm (1)
3	June 25, 1979	bb	Ovality + SCC	135 gpm (1)
4	Oct. 2, 1979	N	Loose part (spring)	390 gpm (1)
5	Jan. 25, 1982	C	Loose part (plate)	634 gpm (2)

Ref.

1. NUREG-0651, Evaluation of Steam Generator Tube Rupture Events, USNRC, Appendices Card H, March 1980.

2. Response to Long Term Commitments, Ginna Restart SER, Steam Generator Tube Rupture Incident, November 22, 1982, Attachment B, Analysis of Plant Response During January 25, 1982, Steam Generator Tube Failure at the R. E. Ginna Nuclear Power Plant.

TABLE 8

SENSITIVITY OF TUBE FAILURE RATE TO NUMBER OF FAILURES EXPERIENCED

Assumed No. of Failures Experienced in 3.6E+06 Tube Years of Operation	Corresponding Failure Rate at Indicated Confidence Level	
	50 percent	95 percent
5	1.6×10^{-6}/Tube Year	2.9×10^{-6}/Tube Year
4	1.2×10^{-6}	2.5×10^{-6}
3	1.0×10^{-6}	2.2×10^{-6}
2	0.74×10^{-6}	1.8×10^{-6}
1.5	0.60×10^{-6}	1.5×10^{-6}
1	0.47×10^{-6}	1.3×10^{-6}
0	0.19×10^{-6}	0.83×10^{-6}

ATTACHMENT A: PRESSURE PULSE MODEL

This exhibit describes the pressure pulse model used to quantify the probability of multiple tube rupture events at the Shearon Harris Nuclear Power Plant.

The 6×10^{-7} per tube-year rupture frequency calculated from the modified experience base is the frequency of degradation to the extent of rupture under the normal operation tube differential pressure load in the range of 1250 psi. The frequency of degradation to the extent of rupture under increased pressure loads is assumed to be of this magnitude also. The model assumes that for a tube that does degrade to this extent, it may take anywhere from 0 to 40 years of operation with equal probability.

For this analysis, transient pressure swings up to the 2500 psia safety valve set point (a pressure differential of 1500 psi) are assumed to occur with a frequency of once per year. The time that a degrading tube spends in the 1500 to 1250 psi capability range is thus estimated to be:

$$t^* = t \left[\frac{L_T - L_{NO}}{L_I - L_{NO}}\right]$$

Where: L_T = the tube capability of a tube failing under a transient load

L_{NO} = the capability of a tube that fails under normal operating loads

L_I = the initial minimum virgin tube burst capability

t = the time for a tube to degrade to 1250 psi capability

This model is shown in Figure A-1.

For the case of a normal transient, L_T is 1500 psi and L_{NO} is 1250 psi (normal operating load). Based upon laboratory testing, the minimum virgin tube burst capability is assessed at 10,000 psi. The time to degrade, t, is assumed to be uniformly distributed from 0 to 40 years of service life. On the average (i.e., the mean time to failure), the time for a tube to degrade would be T/2, or 20 years.

Thus, for this case

$$t^* = [\frac{1500 - 1250}{10,000 - 1250}] \ t = .029t$$

This model does not presume a great level of detail regarding the shape of the tube degradation curve. Although a variety of convex or concave degradation curve shapes are theoretically possible (provided that the tube capability monotonically decreases), a uniform linear rate was used in this model to provide some average sense that the time a failing tube spent in any given strength band is proportional to the width of the band.

Given a transient event, the probability that a tube exposed to a 1500 psi differential pressure would rupture is

1. $p = \lambda t^* = .029 \ \lambda t$ per tube

A weighted average of t* is calculated, yielding a value of 0.59. Thus,

$$p = \lambda t^* = 0.59\lambda$$

The transient pressure differential is applied to all three steam generators. Based on this and the assumption that each tube's failure probability is random and independent, the probability of various numbers of tubes rupturing can be evaluated from the binomial distribution.

2. $$\underline{P}(r) = \frac{n!}{r!(n-r)!} p^r \times (1-p)^{n-r}$$

Where:
- n = number of steam generator tubes = 4578 x 3 = 13,734
- r = number of tubes rupturing, i.e., 1 or 2 or 3
- p = probability of individual tube failure from Eq. 1
- $\underline{P}(r)$ = probability of r tubes failing.

To account for the dependence between steam generator tubes, the method of discrete probability distributions (DPDs) was used to quantify P^r in the above expression. The DPD method is useful when analyzing components of the same type (e.g., steam generator tubes) which have identical probability distributions (or pdfs). These pdfs are not only identical, they are dependent in the sense that, if one were somehow to learn the true failure rate of component 1, this would certainly affect the state of knowledge about the failure rate of component 2. Note, however, that this does not mean that one would know the failure rate of component 2 exactly because, although it is the same type of component, it is physically distinct. The DPD for the second component, however, would be narrower.

A probability distribution for λ was assigned as follows. The five plants which have had tube rupture events make up about 10 percent of the tube experience base. The experienced tube rupture frequency for these "worst" plants (1.5 events/3.6 x 10^5 tube-years = 4.2 x 10^{-6} events/tube-year) is assigned a probability of 10 percent. The median value calculated above was assigned a probability of 80 percent; the lower tail, from the Chi-square tables, was assigned a 10 percent probability. The following distribution is thus assigned for λ:

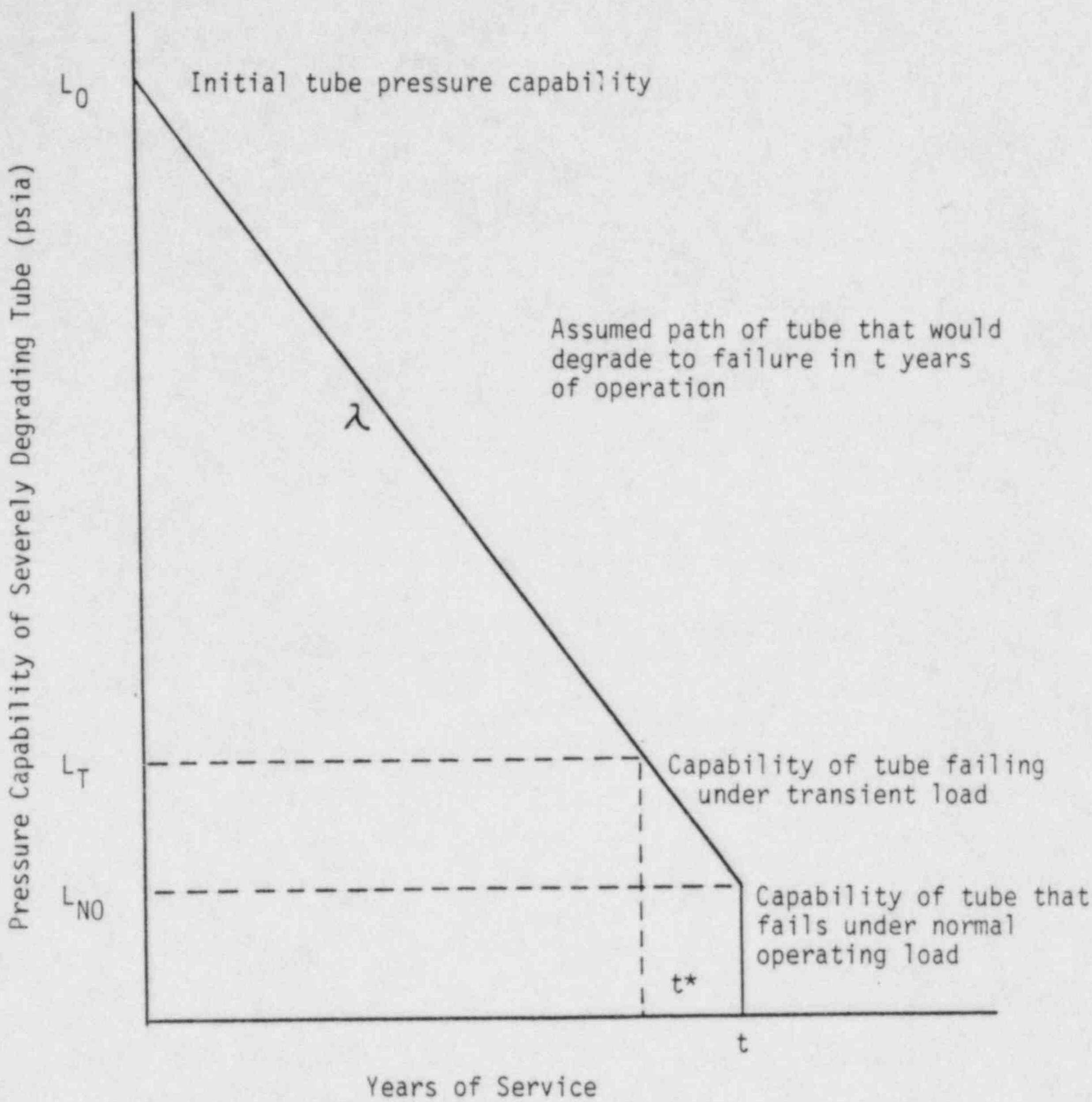
Probability	λ
.1	4.2×10^{-6}
.8	6.0×10^{-7}
.1	1.6×10^{-7}

This model gives the results listed below for rupture of one, two, or three tubes. Since the frequency of these transients has been presumed to be once per year, these probabilities also constitute annual frequencies. These results show a multiple tube rupture frequency of 7 x 10^{-5} per year.

Number of Tubes Rupturing	Probability
1	7.5×10^{-3}
2	6.7×10^{-5}
3	6.7×10^{-7}

FIGURE A-1

MODEL FOR PROBABILITY OF TUBE RUPTURE ON LOAD INCREASE



λ = Frequency of severe degradation or rupture (per tube year)

t = Time to fail under normal load (assumed random over period 0 to 40 years

t* = Time vulnerable to credible steam break load (years)

$$t^* = t\left[\frac{L_T - L_{NO}}{L_0 - L_{NO}}\right]$$

P = Probability of failure given steam break loads = λt^*

$$= \lambda t\left[\frac{L_T - L_{NO}}{L_0 - L_{NO}}\right]$$