UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
LONG ISLAND LIGHTING COMPANY	)	Docket No. 50-322 (OL)
	)	
(Shoreham Nuclear Power Station, Unit 1)	)	
	)	

TESTIMONY OF RICHARD A. HILL
FOR THE LONG ISLAND LIGHTING COMPANY
ON SOC CONTENTION 16 -- CLADDING SWELLING AND FLOW BLOCKAGE

PURPOSE

This testimony shows that the issues raised in NUREG-0630 do not have any adverse impact on the ECCS analysis done for Shoreham. Much of the information presented by the NRC Staff in NUREG-0630 is not applicable to BWR's. GE also conducted sensitivity studies which demonstrate the adequacy of the results of GE's ECCS analysis.

The testimony also shows that the fission gas model issue raised in the contention does not affect the ECCS analysis for Shoreham's first operating cycle. Furthermore, analysis of the impact of improved ECCS models currently awaiting NRC approval shows that the models will result in a substantial reduction in the calculated peak cladding temperature (PCT) for subsequent

cycles. Thus, there is no need to conduct a Shoreham specific ECCS re-evaluation after the first cycle.

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1. Q. Please state your name and business address.

A. My name is Richard A. Hill; my business address is the General Electric Company, 175 Curtner Avenue, San Jose, California.

2. Q. What is your position with the General Electric Company?

A. I am the Manager of Systems Evaluation Programs in the Safety and Licensing Operation for the General Electric Company.

3. Q. Please state your professional qualifications.

A. The resume on pages 21-22 of this testimony summarizes my professional qualifications. My familiarity with the issues raised in SOC Contention 16 stems from work in my current position. I am responsible for resolution of generic technical issues regarding ECCS performance and conformance to regulations.

4. Q. Are you familiar with SOC Contention 16?

A. Yes.

5. Q. What does this contention involve?

A. SOC Contention 16 focuses on the cladding swelling issues raised in NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis."

6. Q. What is cladding swelling and flow blockage?

A. If a loss of coolant accident (LOCA) were to occur, the reactor coolant pressure might drop below the internal fuel rod gas pressure. This pressure differential could cause the fuel cladding to swell and, possibly, rupture. The time at which the swelling and rupture occur and the magnitude of the swelling would affect core conditions during the LOCA. These phenomena are incorporated into GE's ECCS analysis. As stated in NUREG-0630, GE does not use a flow blockage

model in its ECCS analysis. As a result, this testimony will only refer to cladding swelling models.

7. Q. Precisely what issues were raised with respect to cladding swelling in NUREG-0630?

A. NUREG-0630 resulted from the NRC Staff's continuing research on fuel cladding behavior under LOCA conditions. Data developed in that research program were reported in NUREG-0630. Although the NRC Staff had approved the GE ECCS model for use, it requested additional information to reconcile the new test results on fuel cladding swelling and rupture with the models used by GE.

Two areas of inquiry were pursued. First, the new data allegedly indicated a substantial underprediction of the incidence of fuel rupture at high pressure differentials (high stress). Figure 44 of NUREG-0630 (shown on page 18 of my testimony) depicts the curves tracking the NRC data. The figure also includes a curve based on GE data (the "GE curve"). These curves represent the temperature at which cladding rupture is expected to occur for a given stress (which is a function of pressure differential). For analytical purposes, the GE and NRC curves can be divided into two areas in which variances occur: (i) the low

temperature (below 870°C), high stress portion of the curves, and (ii) the high temperature, low stress portion of the curves (above 870°C). As indicated in NUREG-0630, the low temperature, high stress portion of the curves is not relevant to the GE ECCS model because BWR fuel pre-pressurization is much less than in PWR's and is not a significant contributor to fuel cladding perforation. However, the variances at the high temperature portion of the curves (above approximately 870°C) still had to be explained. By separate correspondence, the NRC Staff asked GE to provide supplemental calculations using the most conservative NRC curve (0°C/sec) from NUREG-0630.

The second issue raised by NUREG-0630 involved the circumferential burst strain versus temperature curves. Burst strain is related to the amount of deformation of the cladding at the location of a rupture (the higher the strain, the greater the expected deformation). The GE and NRC curves are depicted on Figures 45 and 46 of NUREG-0630 (see pages 19-20). As noted by the NRC Staff, the fast heat-up curves in Figure 46 are not applicable to BWR's. The area of concern for BWR's was "[f]or temperatures above 925°C and for slow ramps." NUREG-0630 at 61. For that portion of the curve, the GE curve underpredicted the NRC

curve. Thus, GE had to provide further information. Also, in separate correspondence, the NRC Staff asked GE to perform calculations using a correlation which bounded a combination of the slow and fast NUREG-0630 heat-up strain curves.

8. Q. With regard to the first issue -- the underprediction of the incidence of fuel rupture -- does the NRC data indicate a deficiency in the GE ECCS model?

A. As already noted, the NRC conceded in NUREG-0630 that the data below approximately 870°C were not applicable to BWR's. Similarly, above 870°C the data supporting the NRC curves are not applicable to the GE ECCS model because the pertinent NRC curves are based on fast heat-up rate data rather than the slow heat-up rate data characteristic of boiling water reactors. The GE curve is based on a considerable amount of slow heat-up rate data accumulated by GE. Thus, the portion of the NRC curves above 870°C should also be disregarded.

In order to further demonstrate the adequacy of the GE ECCS model, GE performed sensitivity studies which compared the current GE stress curve with a modified stress curve (NUREG-0630 curve below 870°C and the GE curve above 870°C). The results showed that using

this modified curve, the impact on the peak cladding temperature would be no more than plus or minus 10°F. Such a variance is not considered significant under the provisions of NRC regulations (Part II, paragraph 1.b. of Appendix K to 10 C.F.R. Part 50).

9. Q. And what was GE's response to the second issue -- the circumferential rupture strain versus temperature curves?

A. The NRC strain versus temperature curves on Figures 45 and 46 are inapplicable to GE BWR fuel. As recognized in NUREG-0630, fast heat-up rate data cannot be applied to BWR's, which have a maximum heat-up rate of less than 10°F/sec. A combination of the slow and fast heat-up rate curves would be similarly inapplicable.

The criteria used to select the data from which the slow heat-up rate curve below 925°C was derived are suspect. NUREG-0630 states that most of the data falling below this curve were discounted because they were derived from tests with features known to reduce perforation strain, e.g., nonuniform temperature profile, corrosion fission products and cold shrouds. All of these features, however, would be present in a BWR during a LOCA. Unless the test conditions

accurately reflect the BWR design, the results will not be meaningful. Since the majority of the NRC data used were obtained under conditions not prototypical of the BWR, the applicability of any correlation derived from these data is questionable.

For temperatures above 925°C, which is the region applicable to BWR's, the GE curve shown in Figure 45 of NUREG-0630 is an average of the localized rupture strain over a 3" axial distance of the fuel rod. The GE curve plotted without the 3" averaging effect (to be consistent with the method by which the NRC curve was generated) is much more conservative than the NUREG-0630 slow heat-up rate curve (see page 21).

In addition to the review of the data just described, GE performed sensitivity studies to determine the effects of the NRC strain versus temperature curves on the GE ECCS model. These studies were performed using a base case plant with characteristics that bound all BWR's including Shoreham. The majority of the studies were performed using pre-pressurized 8x8 fuel. The temperature versus rupture strain input to the GE ECCS model was varied to determine its effect. In other words, assuming that higher rupture strain might occur as suggested by the NRC, GE wanted to determine the

impact on peak cladding temperature (PCT). The results of the sensitivity studies showed decreases in PCT (as much as 40°F) with the higher rupture strains. This reduction is due mainly to the increased heat transfer area available on the fuel cladding at the higher strains.

10. Q. Mr. Hill, would you please summarize your conclusion about the NUREG-0630 issues?

A. The NRC curves in NUREG-0630 (Figures 44-46, pages 18-20 below) do not affect the adequacy of the GE ECCS model nor the accuracy of the underlying calculations required by NRC regulations in Appendix K to 10 CFR, Part 50. This conclusion is based on the analyses of the relevant data and the sensitivity studies performed by GE.

11. Q. Has GE responded to the NRC Staff on NUREG-0630?

A. Yes. GE's position has been explained to the Staff in submittals and in meetings.

12. Q. And what has been the Staff's response?

A. The Staff has orally concurred with GE's conclusions. A generic Safety Evaluation Report on the issue is expected soon. In the case of Shoreham, acceptance

was indicated in Supplement No. 1 to the Shoreham SER on page 4-1. The Staff did, however, impose a license condition requiring ECCS reanalysis for the second fuel cycle and beyond, "utilizing models that (a) account for the effects of high-burnup fission gas release and pre-pressurized fuel, (b) accomodate the information in NUREG-0630 including its effects on local oxidation, and (c) have been reviewed and approved by the NRC."

13. Q. Why did the Staff impose this condition?

A. When the NRC Staff completed its review of NUREG-0630, it stated, in Supplement No. 1 to the Shoreham SER (pages 4-1 to 4-2), that LILCO had submitted information to resolve the NUREG-0630 issues (as well as information on a separate issue regarding a GE fission gas model) for Shoreham. The Staff, however, felt there were some uncertainties in the information. Instead of closing the issue, it proposed the above license condition citing seven factors as the reasons. It is precisely those seven factors that SOC cites as the reasons why Shoreham has not adequately considered clad swelling and flow blockage.

14. Q. Before we get to a discussion of these seven factors, you mentioned an issue regarding GE's fission gas model. Please explain.

A. The fission gas model issue was not raised in NUREG-0630. When SOC submitted this contention listing the factors cited by the Staff in the Shoreham SER, it incorrectly linked it to NUREG-0630. I will, however, address the issue in this testimony.

Fission gas released due to the fissioning of the U-235 in the fuel pellets causes the pressure inside the zircalloy rods to increase over the lifetime of the fuel. GE's current fission gas release (FGR) model used with the ECCS evaluation model adequately predicts the FGR up to fuel burnups of 20,000 mwd/STU. This burnup is reached sometime beyond the first fuel cycle. After that point the GE model begins to underpredict the release. The NRC has requested that GE either use an NRC correction factor with the current FGR model or submit a completely new model. GE submitted a new model in December 1981 and NRC approval of the model is expected by December 1982. I will discuss this improved model later on in this testimony.

15. Q. Let's get back to issues raised in the contention. What are the factors SOC cites in support of its argument that clad swelling and flow blockage problems have not been resolved for Shoreham?

A. SOC alleges seven reasons in SOC Contention 16(a) that can be summarized as follows:

(i) a lack of margin in the calculated LOCA peak clad temperatures (PCT);

(ii) the use of data for unpressurized fuel;

(iii) the incomplete nature of the analysis of enhanced fission gas release;

(iv) the uncertainties associated with calculating the net change in PCT resulting from use of the new ECCS models;

(v) the preliminary nature of the PCT results using the new ECCS models;

(vi) the failure to account for the effects of zircalloy oxidation heat; and

(vii) the failure to use "base case flow blockage" in the burst-strain sensitivity study.

In response to LILCO's interrogatories ("Response of SOC to LILCO Discovery Request dated February 23, 1982," dated March 17, 1982), SOC indicated that it is no longer interested in pursuing the last of these issues, so I will not address it here.

16. Q. With respect to the six remaining issues, some appear to be related. Would it be easier to address them if they were grouped together?

A. Yes. I would like to address items (i),(ii) and (vi) individually. Items (iii), (iv) and (v) are all related to the fission gas model and I will address them together.

17. Q. With regard to SOC Contention 16(a)(i), is there any margin to the 2200°F PCT limit for Shoreham?

A. Using the current GE ECCS model approved for use by the NRC, the calculated LOCA peak cladding temperature for Shoreham is 2200°F. This does not mean, however, that there will be no margin to the LOCA PCT limit during operation of Shoreham. The evaluation model predicts 2200°F PCT only at the most limiting time in the operating cycle. The calculated PCT at other times in the cycle shows a margin to the PCT limit. Also, the Appendix K ECCS evaluation models are extremely conservative. Test data and more realistic analyses have shown that the actual PCT never exceeds approximately 1200°F. Thus, realistically, there is a considerable margin to PCT.

Furthermore, as I have already explained, there is nothing in NUREG-0630 that would have any significant adverse affect on the calculated PCT using the currently approved conservative models. With regard to the stress versus temperature curve, it was shown that using a curve bounding the appropriate NRC data, the overall PCT impact was $\pm$ 10°F. This change is insignificant.

With regard to the strain versus temperature curves, I explained why GE believes the new NRC data may not be applicable to BWRs. Again, GE performed sensitivity studies to determine the effects of the NRC strain versus temperature curves on the GE ECCS model. The results showed a decrease in PCT as much as 40°F with the higher rupture strains. The reduction is due primarily to the increased heat transfer area available as the fuel cladding swells.

18. Q. SOC Contention 16(a)(ii) questions the use of data derived from unpressurized fuel. Is that a valid concern?

A. No. Shoreham will use pre-pressurized fuel and General Electric has conducted a study that shows that use of pre-pressurized fuel in BWR's reduces the calculated PCT. This study has been submitted to and approved by the NRC Staff. The current maximum calculated PCT overpredicts the PCT calculated assuming pre-pressurized fuel by as much as 60°F.

19. Q. SOC Contention 16(a)(vi) alleges that the GE LOCA analysis has not accounted for zircalloy oxidation heat. Is this true?

A. No. Zircalloy heating oxidation has always been accounted for in the current GE LOCA model. Furthermore, GE sensitivity studies have shown that any increased oxidation heat generated at higher strains is offset by improved heat removal from the rods due to larger surface area.

20. Q. Let's go to the question of the fission gas model and SOC Contentions 16(a)(iii), (iv) and (v). Please explain whether there is any validity to these criticisms.

A. As explained before, the NRC Staff has raised questions about the adequacy of GE's fission gas model at high fuel burnup. GE has been working to improve not only this model but its complete fuel performance model and its ECCS model. In December 1981, GE submitted an improved fuel performance model (which includes a fission gas model) and an improved ECCS evaluation model to the NRC for approval, which is expected by December 1982. GE has analyzed the impact of the new ECCS and fuel performance models on calculated PCT's. We have concluded that their use will result in a very substantial reduction in the maximum calculated PCT. For a plant such as Shoreham, the maximum PCT will drop by approximately 500°F to 1000°F.

The ECCS calculations for Shoreham were done using NRC approved ECCS and fuel performance models. The calculated PCT was within the prescribed limit of 2200°F. GE's analysis shows that use of the new, more realistic models, including an improved fission gas model, will substantially reduce the calculated PCT for Shoreham. In my view, there is no need to conduct a Shoreham-specific reanalysis to find out precisely how much of a reduction will occur.

21. Q. Despite the NRC Staff's reservations about the current fission gas model, it still concluded that operation during the first cycle was acceptable. What was the basis for that conclusion?

A. The NRC Staff concluded that operation during the first fuel cycle would be acceptable because any uncertainties in fission gas effects would only occur at high fuel burnup. For the low burnups that would be experienced in the first cycle, fission gas effects are well known and adequately taken into account.

22. Q. Do you agree that uncertainties in fission gas effects, if any, would not be applicable during the first fuel cycle?

A. Yes.

23. Q. Turning to SOC Contention 16(b), what issue is raised there?

A. SOC alleges that there is inadequate assurance that the reanalysis requested by the Staff in the SER will show compliance with the applicable regulations.

24. Q. And what is your response?

A. As I have already explained in this testimony, the ECCS analysis performed for Shoreham, and the results obtained, meet the requirements of 10 CFR § 50.46 and Part 50, Appendix K for the issues raised in this contention. In addition, GE has already submitted an improved analysis package to the NRC for approval. Use of these new models will substantially reduce calculated PCT's. These new models take into account the information discussed earlier on cladding swelling and fission gas release.

25. Q. Mr. Hill, would you please summarize your conclusions.

A. The ECCS analysis for Shoreham has adequately taken into account cladding swelling and fission gas release. None of the concerns raised by SOC have any impact on that conclusion. Moreover, I believe the

licensing condition proposed for Shoreham by the NRC Staff is unnecessary.

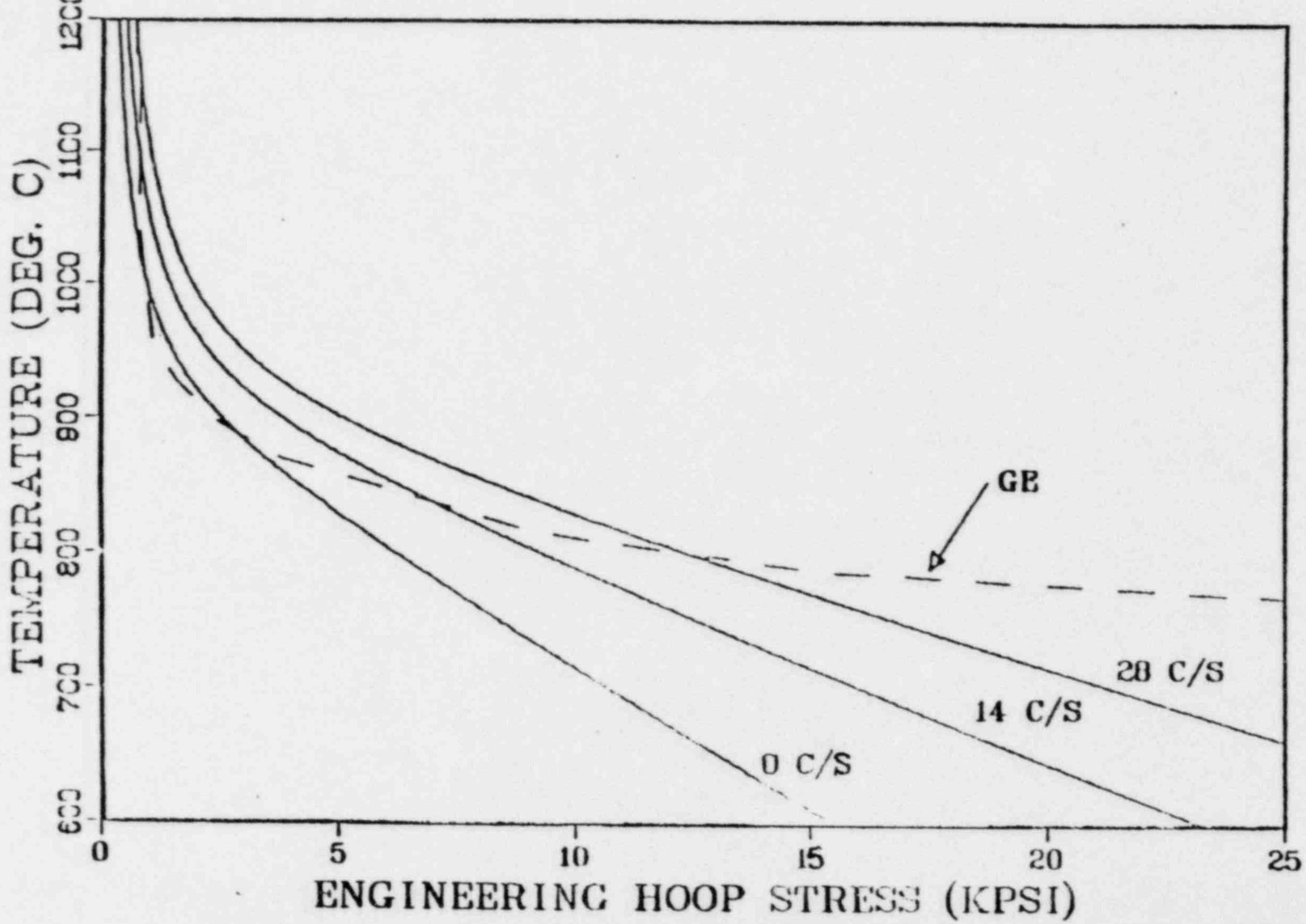

Fig. 44 GE model and ORNL correlation of rupture temperature as a function of engineering hoop stress and ramp rate.

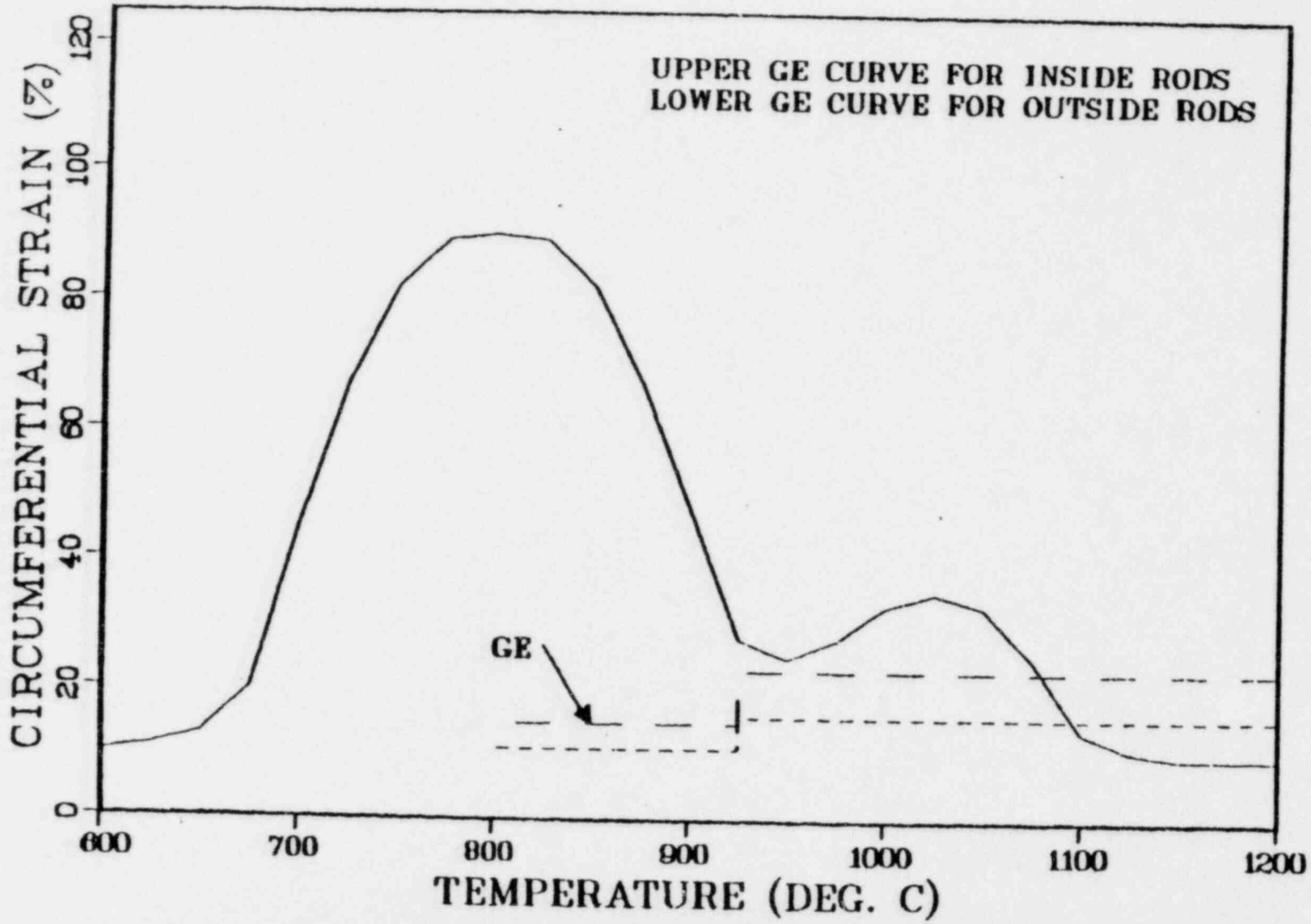

Fig. 45 GE model and slow-ramp correlation of circumferential burst strain as a function of rupture temperature.

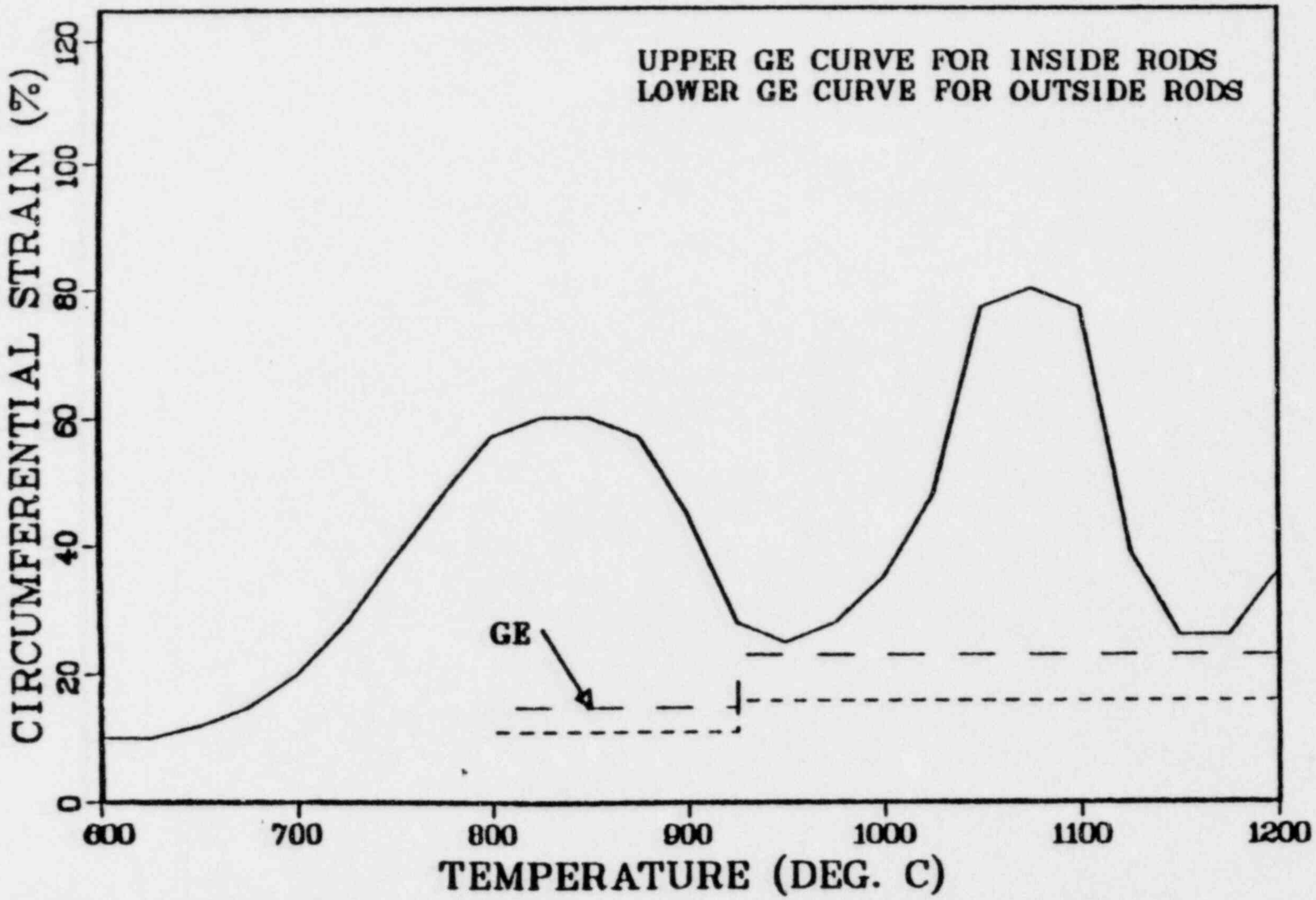

Fig. 46 GE model and fast-ramp correlation of circumferential burst strain as a function of rupture temperature.

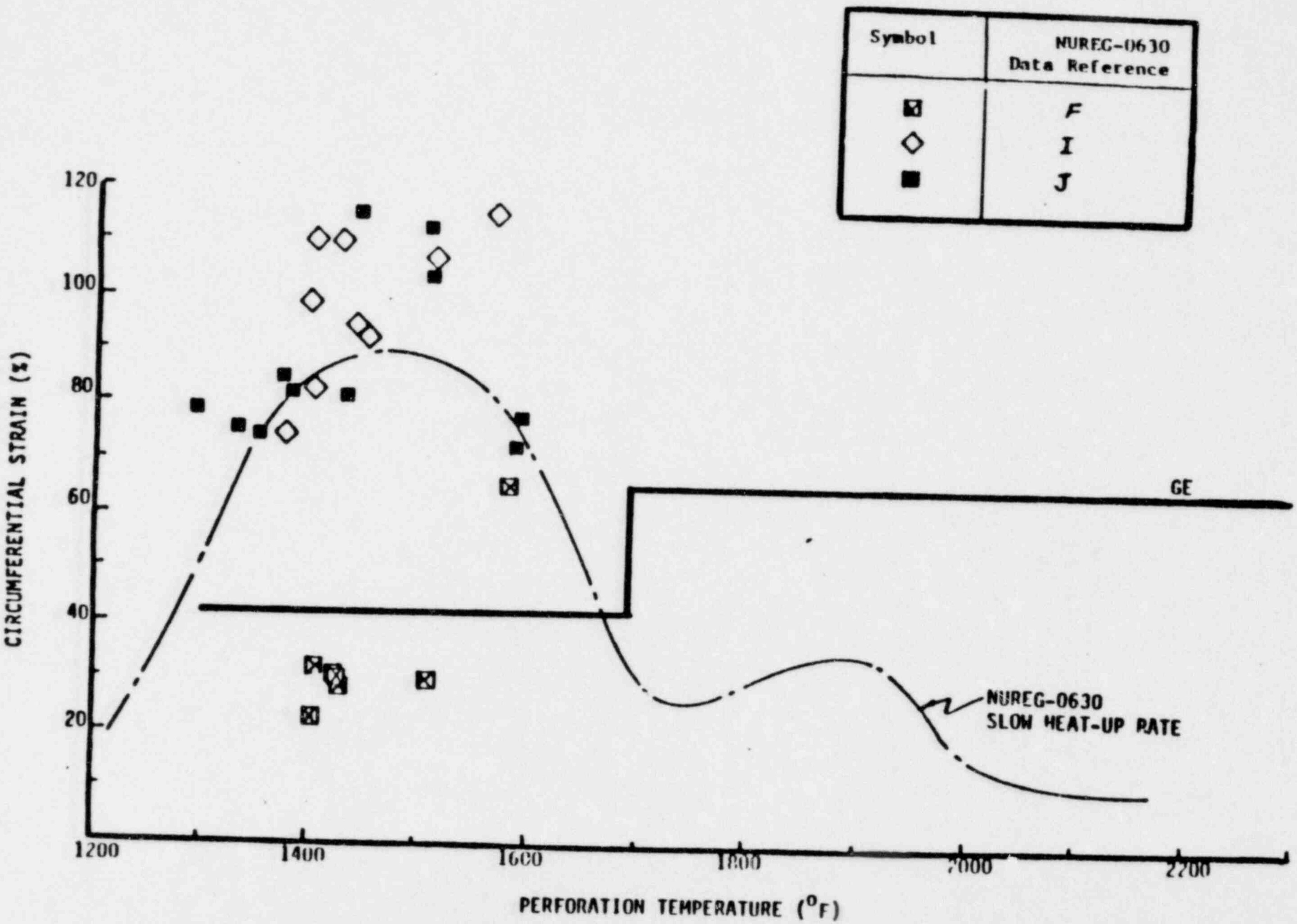

FIGURE 9: NUREG-0630 DATA FOR HEAT-UP RATES $\leq 10^oF/SEC$

Richard A. Hill
Systems Evaluation Programs Manager
General Electric Company

My name is Richard Hill. My business address is 175 Curtner Avenue, San Jose, California. I am employed by General Electric Company (GE) as Systems Evaluation Programs Manager and have held this position since September 1980. In this capacity, I supervise technical program managers for several licensing issue topics.

I received a Bachelor of Arts in biochemistry from the University of California at Berkley in 1969, and a Master of Science in engineering management from the University of Pittsburg in 1977. I have also completed a continuing education course in reliability and risk analysis at George Washington University, and one in man-machine interface engineering at the University of Wisconsin.

Following five years' service in the United States Navy nuclear power program, I joined Westinghouse Electric Corporation, where I was Senior Engineer in the Westinghouse Pressurized Water Reactor Systems Division (1974-1977). In that capacity I acted as program manager and was responsible for planning, implementing, and controlling multi-divisional research programs in human factors and systems integration.

I moved to GE in 1977. From 1977 to 1980 I was Principal Engineer acting as program manager responsible for coordination and integration of programs in dynamic load analysis of equipment and BWR safety analyses in response to Three Mile Island. I became Systems Evaluation Program Manager in September, 1980.