UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

HOUSTON LIGHTING & POWER COMPANY	)	
	)	
	)	Docket No. 50-466
	)	
	)	
(Allens Creek Nuclear Generating	)	
Station, Unit No. 1)	)	

AFFIDAVIT OF MARTIN R. TORRES

State of California
County of Santa Clara

I, Martin R. Torres, Manager, Flow Induced Vibrations, Nuclear Power Systems Engineering Department of the General Electric Company, of lawful age, being first duly sworn, upon my oath certify that the statements contained in the attached pages and accompanying exhibits are true and correct to the best of my knowledge and belief.

Executed as San Jose, California
July 29, 1980.

Martin R. Torres

Subscribed and sworn to before me this 29th day of July, 1980.

Ruthe M. Kinnamon
NOTARY PUBLIC IN AND FOR SAID
COUNTY AND STATE

My commission expires March 28 of 1981

OFFICIAL SEAL
RUTHE M. KINNAMON
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My comm. expires MAR 28, 1981

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(Allens Creek Nuclear Generating Station, Unit No. 1)	§	
	§	
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Affidavit of Martin R. Torres

My name is Martin R. Torres. I am employed by the General Electric Company as Manager, Flow-Induced Vibration, Nuclear Systems Engineering Department. I have served in this capacity for five years. A statement of my experience and qualifications is set out in Attachment 1.

This affidavit addresses TexPirg Contention 11 and Doherty Contention 31. These Contentions state that flow-induced vibrations on the following components have not been adequately assessed for ACNGS:

(a) Jet Pumps

(b) Spargers

(c) Fuel pins

(d) Fuel rods

(e) Incore instrumentation

(f) LPRMs

The phenomenon of flow-induced vibrations has been studied extensively on previous General Electric plants. Information gathered from these studies and test programs designed specifically to study flow-induced vibration has been used to improve design and to qualify ACNGS. Four sets of analyses and tests verify that flow-induced vibration will not impair the safety of ACNGS. These are:

1. A dynamic system analysis.
2. Flow tests, forced oscillation tests, and other physical tests of reactor internal components.
3. Prototype plant pre-operational and operational tests.
4. Pre-operational tests of ACNGS.

The dynamic system analysis is described in Section 3.9.1.3 of GESSAR 238 NSSS and has been in use since the licensing of Browns Ferry Unit 1. This analysis described flow-induced vibration which may result from normal reactor operation. This analysis serves two functions. GE uses it during the design and in-house testing phase of reactor internal components. The dynamic system analysis is also used to establish criteria for plant pre-operational vibration testing (Item 3 above). For example, such an analysis has been prepared for Perry Station Unit 1, the prototype 238 BWR-6 plant. This analysis is fully applicable to ACNGS.

Flow tests of various reactor internals to quantify flow-induced vibration levels are an integral part of General Electric's design process. These tests are conducted to verify design and are independent of vibration testing required by NRC regulations. The tests were performed at various test facilities starting as early as 1974 for BWR-6. The tests followed procedures required by 10 CFR 50, Appendix B. In most instances, these tests were performed using full scale, actual reactor hardware at flow rates well in excess of the operational design condition. Tests included both flow tests and, as appropriate, forced oscillation tests. Flow tests were performed on jet pumps, control rod guide tubes, low-pressure coolant injection lines, feedwater spargers, fuel assembly, in-core instrument tubes, and differential pressure lines and other components.

In one test facility, the High Flow Hydraulic Facility located at General Electric's Nuclear Energy Division in San Jose, California, a full scale mock-up of a segment of the BWR-6 core and lower plenum was flow tested to verify that flow-induced vibration amplitudes were within acceptable levels. Other components, including in-core tubes containing instrumentation such as the LPRMs, fuel bundles and feedwater spargers were tested for flow-induced vibration in various other test facilities. As an example, the feedwater sparger was flow tested at GE's Feedwater Sparger Test Facility in

San Jose. Fuel and in-core instrument tubes were flow tested in the Building G, Large Tank Hydraulic Flow Loop in San Jose. At the Pacific Gas and Electric Company's fossil power plant at Moss Landing, such components as jet pumps were tested with steam and water at BWR operating conditions in a General Electric test facility. Additional testing of various components was done at the Colorado State University Hydraulic Laboratory, Fort Collins, Colorado, (one-fourth scale model of BWR-6) and the Atlas Test Facility, San Jose, California.

The vibration testing requirement of Regulatory Guide 1.20[1/] will be satisfied on a prototype plant, presently designated as Perry Unit 1, whose operation is expected to precede that of ACNGS. On the prototype plant, extensive vibration measurements will be made on major internal components, including the jet pumps, during pre-operational and start-up flow testing, and an extended pre-operational flow test and inspection will detect evidence of possible undesirable effects due to vibration.[2/] Vibratory responses will be recorded at various recirculation flow rates and power levels, using strain gauges, accelerometers and linear differential transducers, as appropriate. Actual results of the data analysis, natural frequencies and mode shapes will then be compared to those obtained from the theoretical dynamic systems analysis discussed above. Vibratory amplitudes will be compared to the criteria derived from that analysis.

For the prototype plant, sensors will monitor possible flow-induced vibration during power operation for the jet pumps, spargers, LPCI coupling, core support structure, fuel channels, LPRMs and other in-core instrumentation. From the fuel channel sensor measurements, information on possible flow-induced vibration effects for the fuel pins and control blades can be derived. These tests will continue until power operating conditions are reached, and are scheduled to be completed prior to operation of ACNGS. In the unlikely event that ACNGS station becomes the prototype plant, the Applicant will follow Regulatory Guide 1.20 and perform these extensive pre-operational and operational tests.[3/]

Because ACNGS is not expected to be the prototype plant for the vibration testing requirement of Regulatory Guide 1.20, confirmatory pre-operational flow-induced vibration testing of reactor internals at ACNGS will be performed in accordance with the "non-prototype" testing provisions of Regulatory Guide 1.20.[4/] The confirmation will be made by the use of extended high-flow testing, preceded and followed by a full inspection of internals in accordance with Regulatory Guide 1.20.

The extensive four-step testing and analysis program described above is fully expected to eliminate flow-induced vibration at ACNGS. Apart from this program, however, at least four other factors provide assurance and protection against flow-induced vibration at Allens Creek:

1. Reactor instrumentation can detect some vibration problems long before they pose any hazard.
2. Design improvements will reduce the possibility of vibration damage to components such as the feedwater sparger.
3. ACNGS will have a loose parts monitoring system.
4. Degradation or failure of some nonsafety components will not prevent a safe shutdown.

Experience has shown that reactor instrumentation can detect some vibration problems long before they pose any hazard. For example, instrumentation such as that measuring differential pressure, jet pump drive flows and pressures, feedwater flows and pressures and neutron sensors have, at times, shown anomolous performance that indicated possible vibration.

At Duane Arnold and Cooper nuclear plants, vibration in LPRM tubes was detected by examination of the transversing in-core probes (TIPs), based on the neutron noise level in unfiltered TIP tracts.[5,6] This instrumentation also led to detection of in-core vibration at Browns Ferry. The ACNGS design, however, eliminates the source of this vibratory

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wear, which was traced to by-pass flow holes in the design of those plants. By plugging these holes and providing an alternate flow path, the problem was eliminated. Bypass flow holes are not a part of the design at ACNGS, so the same problem cannot occur.

Design improvements have produced components less likely to be damaged as a result of flow-induced vibration. For example, an improved interference fit feedwater sparger design will be employed at ACNGS. The design consists of three concentric thermal sleeves, ensuring that detrimental vibration is eliminated under all conditions.

ACNGS will have a loose parts monitoring system, an acoustic system designed specifically to detect any loose parts in the reactor.[7/]

Degradation or failure of some nonsafety components such as feedwater spargers will not affect the capability of the plant to achieve and maintain a safe shutdown condition. In any event, should a feedwater sparger become inoperative, a redistribution of inlet water temperature, flow rate, or in-core neutron flux would be detected by existing in-core instrumentation, allowing corrective action to be taken.

In conclusion, the lesson of nuclear plant operating history is that neither a loss of plant safety nor an inability to safety shutdown the plant has ever occurred because of flow-induced vibration. Moreover, all modifications to operating plants which have proven satisfactory in preventing flow-induced vibration have been rigorously implemented in the BWR-6 design.

References

1/ Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Pre-operational and Initial Startup Testing," Rev. 2, May 1976.

2/ Letter, G. G. Sherwood (GE) to E. G. Case (NRC), "Reactor Internals Vibration Assurance Program," MFN/169/78, April 24, 1978.

3/ PSAR, Appendix C, p. C1.20-1.

4/ GESSAR-238 NSSS, Section 4.2.2.4.

5/ Letter, K. Goller (NRC) to D. G. Eisenhut (NRC), "Modification to Eliminate Significant Incore Vibration," dated March 2, 1976.

6/ Safety Evaluation Report by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, in the Matter of BWR Channel Box Wear, July 22, 1975.

7/ PSAR, Section 1.3.1.2.1.

ATTACHMENT 1

Martin R. Torres

EDUCATION: BSME, 1965, University of Utah
GE Advanced Engineering Course Graduate, 1965-1968
MSME, 1968, University of California, Berkeley
*, Applied Mechanics, Stanford University

EXPERIENCE:

Present Position: Manager, Flow Induced Vibrations, Nuclear Power Systems Engineering Department, General Electric Company

1975-Present: Manager, Flow Induced Vibrations
Responsibilities include management of engineers and technicians engaged in highly technical and experimental work in the field of flow induced vibration. Mr. Torres is responsible for the flow induced vibration testing of all BWR reactor internals. The world's largest test facility - the High Flow Hydraulic Facility - is under his active direction. He was task force leader in several key FIV test programs, such as the feedwater sparger and incore instrument tube. He carries management responsibility for the extensive FIV Program for Light Water Reactors funded by the U.S. Department of Energy from 1976 to the present. Reporting to him on this program are the FIV technical teams of Argonne National Laboratory and General Electric's Corporate Research and Development Center as well as the NPSED Program Manager, Dr. Mark A. DeCoster. In summary, Mr. Torres has performed or directed every General Electric FIV experiment relative to the BWR since 1972.

1972-1975: Senior Development Engineer
Primarily responsible for BWR flow induced vibration (FIV) test programs. Plan, coordinate design hardware/instrumentation, data acquisition and analysis for FIV test program. Extensive internal reports written in the area of flow induced vibration development testing:

1. Feedwater Sparger Vibration Testing
2. Jet Pump Vibration Testing
3. Incore Instrument Tube Fuel Channel Testing
4. Scaling and Model FIV Testing
5. FIV of Cylindreical Rods in Parallel Flow
6. FIV of Inclined Cylindrical Rods in Longitudinal Flow

1968-1972: Dynamic Analysis Engineer - BWRSD
Seismic and vibration analysis of nuclear power plant equipment. Brief list of analytical work performed:

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*Extensive graduate work undertaken from 1969-1979 at Stanford, Fluid Mechanics, Dynamics, Material Science, Solid Mechanics and Mathematics (~100 graduate quarter hours).

1. Equivalent Damping Under Random Vibration
2. Recirc Loop Vibration Analysis
3. Dynamic Loads Due to LOCA
4. Seismic Analysis of RPV and Internals
5. Nonlinear Analysis of CRD Housings
6. FIV Analysis of BWR Componenets
7. Probabilistic Approach to Seismic Analysis
8. Fatigue Useage of BWR Internals

Several papers published in ASME Transactions and earthquake engineering literature on above subjects.

1965-1967: NED Engineering Rotation Program
Thermal-Hydraulics - Transient Analysis BWR
Thermal-Hydraulics - Fast Steam Cooled Reactor
Stress Analysis Engineer - Fast Flux Test Facility
Product Design Engineer - Nuclear Instrumentation Department

Since 1969, Mr. Torres has been a lecturer in dynamics and mechanical vibrations for General Electric's (internal) Advance Engineering Program. Mr. Torres is a registered California Professional Engineer (Lic. No. 14674), a Member of ASME and the following Honor Societies:

Pi Tau Sigma
Tau Beta Pi
Phi Kappa Phi

Magna Cum Laude, 1965 BSME

Martin R. Torres 7/30/80