UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
PENNSYLVANIA POWER & LIGHT COMPANY	)	
	)	Docket Nos. 50-387
and	)	50-388
	)	
ALLEGHENY ELECTRIC COOPERATIVE INC.	)	
	)	
(Susquehanna Steam Electric Station,	)	
Units 1 and 2).	)	

AFFIDAVIT OF JOHN F. COPELAND AND FRANKLIN E. COOKE
IN SUPPORT OF SUMMARY DISPOSITION
OF CONTENTION 8

County of Santa Clara	)	
	:	ss.
State of California	)	

John F. Copeland and Franklin E. Cooke, being duly sworn according to law, deposes and says:

1. As Senior Engineer and Principal Engineer, respectively with General Electric Company ("GE") we give this affidavit in support of Applicants' Motion For Summary Disposition of Contention 8. We have personal knowledge of the matters set forth herein and believe them to be true and correct. A summary of our professional qualifications and experience is attached as Exhibit "A" and "B" hereto. Mr. Copeland is responsible for those portions of this affidavit which pertain to the

toughness properties of materials in the reactor vessel. Mr. Cooke sponsors those portions of this affidavit which discuss the use of materials properties in setting design specifications.

2. The purpose of this affidavit is to respond to Contention 8 in this proceeding:

> The Applicants have not adequately demonstrated compliance with the Standard Review Plan, §5.3.6, "Reactor Vessel Integrity", Part II.6. As a result, the reactor pressure vessel may not survive the thermal shock of cool ECCS water after blowdown without cracking.

3. At the start, it must be noted that compliance with the Standard Review Plan is not required. As stated in the legend accompanying §5.3.3 (and other sections) of the Standard Review Plan:

> Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required.

In any case, the reactor pressure vessels for the Susquehanna Steam Electric Station ("SSES") do meet the thermal shock criterion specified in §5.3.3, Part II.6. That criterion states:

> Abnormal operational occurrences must not result in loss of reactor vessel integrity. The most severe postulated transient is the thermal shock to the vessel caused by emergency core cooling system operation after a loss-of-coolant accident. The criterion for acceptable behavior is that the vessel must remain leaktight enough to support adequate core cooling. The generally accepted principles and procedures of linear elastic fracture mechanics provide the bases for acceptance of analyses that support conformance with this criterion.

Analyses performed by General Electric for the SSES reactor vessels demonstrates that they will survive, with considerable margin, the thermal shock of cool ECCS water after blowdown without cracking. These analyses were performed in accordance with the ASME Boiler and Pressure Vessel Code.

4. To assure that the reactor vessels will not crack in this situation, it is necessary to determine that the particular combination of stress intensity and temperature needed to cause crack propagation does not occur.

5. When cool ECCS water reaches the reactor vessel, stresses are created in the vessel wall because the inside surfaces cool faster than the outside surfaces. These stresses are calculated using a finite element program and standard mechanical engineering procedures and the highest stress location is determined. As required by the Code, a one-quarter through wall flaw is then postulated at this location (even though the existence of such a flaw is extraordinarily conservative). The postulated flaw serves to concentrate the stresses at that location. Using multipliers specified in the ASME Code, Section 3 Appendix G, Winter of 1974 (and subsequent) addenda, these stresses are translated into a stress intensity factor representing the concentration of the stress at the location of the flaw. The minimum reference nil ductility temperature (minimum RT_{NDT}) at that stress intensity level and material temperature is then determined using a curve specified in the ASME Code, Section 3 Appendix G. This curve is based on many years of test values reported in Welding Research Council Bulletin 175.

Finally, the RT_{NDT} for the reactor vessel material at this critical location is ascertained. If this material RT_{NDT} is lower than the minimum RT_{NDT} the postulated flaw will not propagate from the stresses of cool ECCS water being injected into the vessel.

The material RT_{NDT} is defined by the ASME Code as the lower of:

1) The "nil ductility transition temperature" (NDT) (the temperature at which a flaw will start to propagate when a material specimen is subjected to dropweight tests) (ASTM Standard E208); or

2) the Charpy V-notch (CVN) transverse orientation 50 ft/lbs transition temperature, minus 60°F (the temperature, minus 60°F, at which fifty ft/lbs energy applied opposite to a V-notch will fracture the transversely oriented specimen) (ASTM Standard E23); or

3) the CVN transverse orientation 35 mils lateral expansion transition temperature, minus 60°F (the temperature, minus 60°F, at which the loading applied opposite to a V-notch will cause 35 mils deformation in the transversely oriented specimen) (ASTM Standard E23).

6. For the Susquehanna reactor vessels, the temperature of the ECCS water when it reaches the vessel wall was conservatively assumed to be 50°F. This is the lowest temperature conceivable for the ECCS water, which will come from sources inside the facility for the duration of the thermal shock condition. (The initial sources of ECCS injection water are the condensate storage tank and the pressure suppression pool). Since these injection sources are all within the plant and since the water would pass through pipes and pumps before reaching the vessel and would impact the core shroud before impacting the vessel wall, the 50°F assumption is very conservative.

7. The stresses on the SSES vessels caused by injection of 50°F ECCS water were determined based upon detailed stress analysis for a reactor vessel fabricated of the same materials as those for SSES, with essentially identical wall thicknesses and similar basic dimensions. This analysis showed that for the SSES vessels, the reactor vessel bottom head was the critical location and that an RT_{NDT} for the bottom head material of 94°F or lower would assure that crack propagation will not occur on ECCS injection.

8. Because the RT_{NDT} concept was not added to the ASME Code until after the SSES vessels were designed, the materials obtained and tested, and fabrication started, test results for the SSES vessel material have had to be correlated to the RT_{NDT} definitions. The CVN transition temperature and NDT results were derived from CVN tests in the SSES vessel materials. Under the then applicable version of the ASME Code, the CVN tests were based on longitudinal orientation and 30 ft/lbs (rather than the current transverse orientation and 50 ft/lbs). The CVN transition temperatures have been adjusted, and NDT values estimated based on toughness data from the SSES and other vessels as well as the results presented in Welding Research Council Bulletin 217, "Properties of Heavy Section Nuclear Reactor Steels", to account for the NDT correlations, the energy level and orientation differences. As adjusted, conservative values for RT_{NDT} of the SSES bottom head material are 34°F for Unit 1 and 20°F for Unit 2, well below the 94°F RT_{NDT} at which crack propogation would be predicted to occur.

9. Although RT_{NDT} for reactor vessel material shifts with neutron irradiation of the material (i.e., the RT_{NDT} increases over the life of the plant), the bottom head material is not subject to sufficiently high neutron fluences for the RT_{NDT} to shift. Other areas of the vessel, such as the beltline, which experience higher neutron fluences are not subjected to thermal shock since ECCS coolant water is not injected into the core beltline region in BWR's such as Susquehanna.

John F. Copeland

John F. Copeland

Franklin E. Cooke

Franklin E. Cooke

Sworn to and subscribed before me

this 14th day of July, 1981

Theresa A. Des Jardin

NOTARY PUBLIC

OFFICIAL S[illegible]L
THERESA A DES JARDIN
NOTARY PUBLIC • CALIFORNIA
SANTA CLARA COUNTY
My comm. expires MAR 1, 1985

LS:ggo/30L
7/13/81

Statement of Qualifications - J.F. Copeland

EDUCATION:

B. S. Degree, Metallurgical Engineering, 1966, Drexel University

M. S. Degree, Metallurgy & Materials Science, 1968, Lehigh University

Ph.D. Degree, Metallurgy & Materials Science, 1973, Lehigh University
(Ph.D. thesis on factors influencing toughness of a low alloy steel)

RELATED EXPERIENCE:

1968-1974: Employed at Lukens Steel Company (supplier of nuclear reactor vessel steel plates) in area of alloy steel research, including the relationship of toughness to microstructure, composition, & heat treatment.

1974-1977: Employed at GE-San Jose in the Fast Breeder Reactor Department in areas including U.S. Government (ERDA) funded program on fracture toughness properties of steam generator materials. (Task Leader).

1977 - Present:
Employed at GE - San Jose in Materials Application Engineering for BWR's in the areas of reactor pressure vessel and piping.

TECHNICAL COMMITTEE MEMBERSHIP AND PUBLICATIONS:

Approximately 12 technical publications in area of pressure vessel and piping materials.

Member ASME Boiler & Pressure Vessel Code Subgroup on strength of Ferrous Metals and Subcommittee on Properties of Metals.

Member ASTM E10.02 on Behavior & Use of Metallic Materials in Nuclear Systems.

Member Metals Properties Council Task Groups on effects of radiation on reactor vessel material toughness.

"EXHIBIT A"

FRANKLIN E. COOKE
PRINCIPAL DESIGN ENGINEER
REACTOR PRESSURE VESSEL & INTERNALS DESIGN UNIT

Education

Engineering, 1946-1947, Oklahoma City University, Oklahoma
BSME, Mechanical Engineering, 1947-1950, Southern Methodist University, Dallas, Texas

Summary

Twenty-six years in the nuclear field including nuclear steam supply system design and specifications. Responsibilities encompass definition of fracture toughness, operating limits; evaluation and review of feedwater nozzle seal leakage and thermal tests; and review of product safety standards and regulatory guides.

Professional Experience

Fourteen years, General Electric Company, Principal Design Engineer. Current responsibilities include definition of fracture toughness operating limits for the reactor vessel. Evaluation of feedwater nozzle field leakage and thermal tests. Definition of standard plant reactor specification. Review of product safety standards and regulatory guides. Performance of New Loads and thermal cycle evaluation. Definition of reactor water level operating limits.

Past responsibilities included definition of reactor assembly operational loading conditions and functional design requirements, design criteria and application limits. Integrating system design requirements into reactor component design requirements for the BWR 6 reactor design. Test on reactor water level gradient, steam separator-dryer, and thin control blade testing. Defined reactor vessel loads and reactor cycle conditions integrated reactor design input to the GE Standard Plant Safety Analysis Report (GESSAR), project safety standards and regulatory guides. Provided reactor design and engineering support for all operating reactor problems such as New Loads, feedwater nozzle repairs, core plate plugging, reactor nozzle, safe end replacement. Also prepared design and test specifications for the replacement feedwater nozzle sparger design. Participated on Loads Combination Task Force. Established a generic basis for operating limits to avoid brittle fracture for operating reactors.

Three years, General Electric Company, Technical Leader. Responsible for nuclear boiler and containment arrangement; reactor auxiliary equipment specification; thermal hydraulic analysis and design criteria for the reactor assembly.

"EXHIBIT B"

Fourteen years, General Electric Company, Engineer. Activities included power plant mechanical design; general performance; nuclear steam supply system arrangement and equipment specification for the reactor recirculation piping, pumps, steam·drum and steam generators. Specific sites worked on during this time include Dresden, KRB, Tarapur, and Nine Mile Point.

FEC:pes/594-5
7/6/81

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CERTIFICATE OF SERVICE

This is to certify that copies of the foregoing Applicants' Motion for Summary Disposition of Contention 8, Statement of Material Facts As To Which There Is No Genuine Issue To Be Heard (Contention 8), and Affidavit of John F. Copeland and Franklin E. Cooke in Support of Summary Disposition of Contention 8, were served by deposit in the United States Mail, First Class, postage prepaid, this 16th day of July, 1981, to all those on the attached Service List.

Jay E. Silberg

Jay E. Silberg

Dated: July 16, 1981.

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SERVICE LIST

Secretary of the Commission
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Administrative Judge James P. Gleason
513 Gilmoure Drive
Silver Spring, Maryland 20901

Mr. Glenn O. Bright
Atomic Safety and Licensing
Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Paul W. Purdom
245 Gulph Hills Road
Radnor, Pennsylvania 19087

Atomic Safety and Licensing
Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Docketing and Service Section
Office of the Secretary
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Judith H. Johnsrud
Co-Director
Environmental Coalition on
Nuclear Power
433 Orlando Avenue
State College, Pennsylvania 16801

Susquehanna Environmental Advocates
c/o Gerald Schultz, Esquire
Post Office Box 1560
Wilkes-Barre, Pennsylvania 18703

Mr. Thomas J. Halligan, Correspondent
The Citizens Against Nuclear Dangers
Post Office Box 5
Scranton, Pennsylvania 18501

Ms. Colleen Marsh
Box 558 A, R. D. #4
Mt. Top, Pennsylvania 18707

Jessica H. Laverty, Esquire
Office of the Executive Legal
Director
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Karin W. Carter, Esquire
Department of Environmental Resources
Commonwealth of Pennsylvania
505 Executive House
Post Office Box 2357
Harrisburg, Pennsylvania 17120

James M. Cutchin, IV, Esquire
Office of the Executive Legal Director
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. Thomas M. Gerusky, Director
Bureau of Radiation Protection
Department of Environmental Resources
Commonwealth of Pennsylvania
Post Office Box 2063
Harrisburg, Pennsylvania 17120

Atomic Safety and Licensing Appeal Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555