February 5, 1985

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF TESTIMONY OF JAMES W. CLIFFORD,
JOSEPH J. BUZY, AND RICHARD J. ECKENRODE

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NRC STAFF TESTIMONY OF JAMES W. CLIFFORD, JOSEPH J. BUZY, AND RICHARD J. ECKENRODE

Q.1. What is your name and occupation?

A.1. (Clifford) My name is James W. Clifford. I am employed as an Operational Safety Engineer (Nuclear) in the Procedures and Systems Review Branch, Division of Human Factors Safety, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.

Q.2. What are your qualifications and experience relevant to your testimony?

A.2. (Clifford) I have a Bachelor of Science degree in Systems Engineering. I have experience in the operation, maintenance, event analysis, and testing of naval nuclear propulsion plants and prototypes. During my employment with the U.S. NRC, I have been involved in numerous evaluations of licensee and applicant emergency operating procedures and procedure programs, including evaluations for licensing and for actual operating events. A further statement of my professional qualifications is attached to this testimony.

Q.3. What is your name and occupation?

A.3. (Eckenrode) My name is Richard J. Eckenrode. I am employed as a Human Factors Engineer in the Human Factors Engineering Branch, Division of Human Factors Safety, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.

Q.4. What are your qualifications and experience relevant to your testimony?

A.4. (Eckenrode) I have a Bachelor of Science degree in Aeronautical Engineering. I have been active in the application of the Human Factors discipline to manned systems since 1960. During my employment by the U.S. NRC, I have participated in numerous evaluations of control room designs and design reviews for applicant and operating reactors. A further statement of my professional qualifications is attached to this testimony.

Q.5. What is your name and occupation?

A.5. (Buzy) My name is Joseph J. Buzy. I am employed as a Senior Reactor Engineer (Training and Assessment) in the Licensee Qualifications Branch, Division of Human Factors Safety, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.

Q.6. What are your qualifications and experience relevant to your testimony?

A.7. (Buzy) I have a Bachelor of Science degree in Marine Engineering. I have over 28 years experience in the design, operation, maintenance, event analysis, and training for military and commercial nuclear power plants, including 17 years as an Operator License Examiner for the U.S. NRC. My current responsibilities include evaluation of training and requalification programs for licensed operators and Shift Advisors. A further statement of my professional qualifications is attached to this testimony.

Q.7. What is the nature of your testimony?

A.7. (All) We are providing testimony to address the question of whether the procedures and training proposed by the licensee will provide additional assurance that the TDI emergency diesel generators (EDGs) will be operated within the specified loading capacity.

Q.8. What part do the procedures and training play in the TDI EDG design issue at Shoreham?

A.8. (All) In response to an NRC staff question, the licensee stated in November 1984, that they were relying on procedures and training (i.e., the operators) to keep from overloading the EDGs above a level identified as a "qualified load" during specified conditions. This qualified load we understood to be 3300KW. The specified conditions were a Loss of Offsite Power (LOOP) or a Loss of Offsite Power in conjunction with a Loss of Coolant Accident (LOOP/LOCA). Without the assurance that operators would keep EDG loading less than 3300KW, the NRC staff could not certify the reliability of the EDGs.

In evaluating the EDGs, the design review resulted in a finding that the EDGs were capable of operating at 3500KW, as indicated in the portion of the testimony provided by the consultants to the NRC staff. Assuming the loads and associated loadings that are identified in the FSAR (Table 8.3.1-1) are accurate, and the reliability of the EDGs is acceptable to at least 3500KW, as determined by the NRC staff and its consultants, the operators are no longer required to keep EDG loading less than 3300KW, and the procedures and training are acceptable to be used, as at other plants, to provide additional assurance that the EDGs will be operated within the loading capacity of the machines.

Q.9. Is there reasonable assurance that the EDGs will be operated within their load capacity?

A.9. (All) Based on the information we have reviewed to date, we have not found reasonable assurance that the EDGs will be operated within their load capacity.

Q.10. Describe the review performed to date.

A.10. (All) In early December 1984, we were asked by our Division of Licensing to evaluate the procedures related to EDG operation. We evaluated the following letters to determine the role the licensee intended for the procedures and training.

a. J. D. Leonard to H. R. Denton, dated July 3, 1984
b. J. D. Leonard to H. R. Denton, dated August 22, 1984
c. J. D. Leonard to H. R. Denton, dated September 11, 1984
d. J. D. Leonard to H. R. Denton, dated November 19, 1984 (SNRC-1104)
e. J. D. Leonard to H. R. Denton, dated November 29, 1984

We received the following procedures during the first week of January 1985:

a. Level Control	SP29.023.01, Rev. 4, dated 12/20/84
b. Loss of Offsite Power	SP29.015.01, Rev. 7, dated 12/20/84
c. Loss of Coolant Accident Coincident with a Loss of Offsite Power	SP29.015.04, Rev. 0, dated 12/20/84

d. Emergency Diesel

Generators SP23.307.01, Rev. 12, dated 12/14/84

e. Main Control Room -

Conduct of Personnel SP21.004.01, Rev. 7, dated 9/27/84

We conducted a review of these procedures for useability and technical accuracy. We had numerous comments on the procedures.

In addition to these procedures, we visited the site January 16-17 to evaluate the location and adequacy of the instrumentation and controls to be used during the execution of the procedures, to obtain information on the training program necessary to complete our evaluation, and to obtain additional procedures that would be used during the assumed LOOP or LOOP/LOCA conditions. The following additional procedures were obtained:

f. Emergency Shutdown SP29.010.01, Rev. 4, dated 8/16/84

g. Loss of Instrument Air SP29.016.01, Rev. 4, dated 10/7/83

Q.11. Describe how the information evaluated has led to your current position.

A.11. (Buzy) The most significant finding was that at the time of our site visit, the training department had not yet started to develop a training program to address the integration of the numerous issues that would have to be addressed to operate the plant with the limitation on EDG loading. We therefore had no basis for evaluating the adequacy of the training, or the bases for the training program.

(Clifford) There were a number of concerns regarding the procedures. In several instances, the procedures would have either directed the operators to take actions that would have overloaded the EDGs, or required the operator to decide between various options, without either specifying the options themselves or providing the criteria for choosing between the options.

(Clifford) The number of procedures that were required to be used by the operators simultaneously raised a concern regarding the manageability of the procedures, and the large number of interrelated actions during their execution.

(Eckenrode and Clifford) There was also a concern that the actions that would have to take place outside the control room to determine if a number of non-safety loads were operating may add an unacceptable level of confusion and delay while the operators were trying to mitigate a LOOP/LOCA event. In addition, no means had been provided to keep track of the loads that were being manipulated.

We are requiring that the specific concerns identified during our review be acceptably addressed by the licensee before we complete our evaluation. These specific concerns are addressed in a Request for Additional Information transmitted from A. L. Schwencer to J. D. Leonard dated February 5, 1985.

PROFESSIONAL QUALIFICATIONS

JAMES WILLIAM CLIFFORD

My name is James William Clifford. I am employed as an Operational Safety Engineer in the Procedures and Systems Review Branch, Division of Human Factors Safety, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. I have held this position since October 1980. I have also been assigned as Acting Section Leader, Section A (Procedures) of the Procedures and Systems Review Branch for the period of March 28, 1983 to September 11, 1983. The Procedures and Systems Review Branch reviews and evaluates licensee programs for the technical, human factors, and operational aspects of nuclear power plant operating and maintenance procedures. I was involved in the pre-licensing audit of emergency operating procedures at five (5) applicants' sites, and have review the emergency operating procedure development programs for eight (8) applicants and operating reactors. These reviews included the evaluation of technical guidelines, operational concerns, and the human factors guidelines to be used in the development and implementation of the emergency operating procedures. I was involved as one of the principal staff reviewers for the human factors aspects of emergency operating procedure generic technical guidelines for B&W and Combustion Engineering Owners Group guidelines, and, through the reviews of procedures for three (3) BWR applicants, assisted in the evaluation of the adequacy of the BWR Owners Group guidelines. I was the principal reviewer for the operational and human factors concerns for the Pressurized Thermal Shock generic issue, including audits of emergency operating procedures for six plants.

From July 1978 to October 1980, I was a naval officer qualified to the equivalent of a shift supervisor at the naval nuclear power prototype at Windsor, CT, where my responsibilities included supervision of plant operations, training of new personnel, and ensuring the continued expertise of experienced personnel. From March 1976 to July 1978 I was a naval officer assigned to a nuclear powered ship, where my responsibilities included safe operation of the ship's nuclear power plant.

I earned a BS degree in Systems Engineering from the U. S. Naval Academy in 1974. During my naval service and my employment with the NRC, I have attended several courses, varying from one week to six months in duration, on plant engineering, human factors, and plant operations. I am previously qualified as Chief Engineer Officer for Naval Nuclear Propulsion Plants.

RICHARD J. ECKENRODE
PROFESSIONAL QUALIFICATIONS
HUMAN FACTORS ENGINEERING BRANCH
DIVISION OF HUMAN FACTORS SAFETY

Since December 1980 when I was hired by the U.S. NRC, I have been assigned to the Human Factors Engineering Branch, Division of Human Factors Safety, Office of Nuclear Reactor Regulation. My initial responsibilities included: (1) participation in the development of NUREG-0700, "Guidelines for Control Room Design Reviews," and (2) participation in the onsite control room design reviews required for operating licenses. Subsequently, I have participated in over 20 control room design reviews, 12 of which I directed. I was a member of the NRC Task Forces which reviewed the steam generator tube rupture event at R. E. Ginna Nuclear Power Plant and the ATWS event at Salem Generating Station.

I have been active in the application of the human factors discipline to manned systems since 1960 and have directed or participated in more than 30 major human factors projects. I am a member of the Human Factors Society.

I hold a Bachelor of Science degree in Aeronautical Engineering from St. Louis University and have completed five NRC sponsored courses in Nuclear Reactor Concepts, Radiation/Contamination Protection, Pressurized Water Reactor Fundamentals, BWR Technology, and PWR Simulation.

From 1963 until joining the U.S. NRC in 1980, I was a Principal Associate with Dunlap and Associates, Inc., of Norwalk, Connecticut. Dunlap and Associates, Inc. is a research and consulting firm in the areas of systems and operations analyses and the behavioral sciences including human factors.

Some of my major projects included:

- Development of human factors guidelines for designing CRT color display formats for a large electrical power distribution control room. Subsequently designed a major portion of the displays.

- Development of a task analysis methodology for determining training requirements and training device requirements and characteristics, as applied to Infantry and Cavalry Fighting Vehicles.

- Conducted human factors and systems analyses resulting in man/machine interface design recommendations, procedures development and training requirements recommendations for the following systems and programs:

  ° Optical lens manufacturing facility
  ° Hemotology laboratory
  ° Navy AEGIS combat system program
  ° Trident submarine missile system
  ° Remotely piloted aircraft
  ° UTTAS and research helicopters
  ° Antisubmarine Warfare attack team trainer
  ° Landing helicopter assault ship

- Chemical/biological warfare protective clothing
- Manned orbital laboratory
- Apollo/Saturn prelaunch checkout system

From 1960 to 1963 I was with the Life Sciences Department of McDonnell Aircraft Corporation. During that time I participated in the human factors analysis and design work on projects Mercury and Gemini and on mechanical ground support equipment for the F4 Tactical Fighter aircraft. I also participated in the Mercury astronaut acceleration training program and gathered human performance data to assist in verifying mission reliability estimates.

JOSEPH J. BUZY

Professional qualifications

Current Position: Systems Engineer (Training & Assessment)
Personnel Qualifications Branch
Division of Human Factors Safety
U.S. Nuclear Regulatory Commission

Education: B.S. Marine Engineering - 1954
U.S. Merchant Marine Academy
Kings Point, N.Y.

Experience:

- Military Service - 1954 - 1956 Served as Damage Control Officer and later Engineering Officer on U.S.S. Hollis APD-86.

- Nuclear - 1956 - 1960: Employed by Bettis Laboratories under contract to the Naval Reactors Program as an operating engineer for the Large Ship Prototype, AIW. I was trained and qualified as Chief Operator on the submarine prototype SIW and assisted in training Navy personnel for SIW and later AIW. I later qualified as Chief Operator on AIW and was assigned as test coordinator during the AIW power escalation program. I was later transferred to Newport News Shipyard as a Bettis Laboratory representative during the construction and start-up testing of the U.S.S. Enterprise. I assisted in initial start-up of two reactor plants on the Enterprise.

1960 - 1963: Employed by the Martin-Marietta Corporation as an operations test engineer for the PM-1 plant. The plant was built for the AEC and Airforce in Baltimore, Maryland, and transported to Sundance, Wyoming. At the site I qualified as Shift Supervisor and was in charge of a combined military crew during the start-up and demonstration phases of the PM-1 plant. I trained and qualified a majority of the military crew who later operated the PM-1 plant.

1963 - 1978: Employed by the AEC as Nuclear Engineer in the Operator Licensing Branch. I was trained and qualified as an operator licensing examiner and responsible for developing and administering written and operating examinations under 10 CFR Part 55 for all types of reactor licensed under 10 CFR 55 and 115. I occasionally directed AEC consultants in development and administration of examinations. In 1970, I was appointed as Section Leader for Power and Research Reactors (P&RR). I trained and supervised several OLB examiners in addition to a group of six to eight consultant examiners. The P&RR section administered examinations at all research and test reactors, Babcock and Wilcox, Combustion Engineering, General Atomics (HTGRs at Peach Bottom and Fort St. Vrain) and the sodium cooled reactors, Fermi I and SEFOR.

Examinations also included use of simulators. The P&RR section occasionally provided personnel to conduct examinations at the Westinghouse and General Electric plants. The P&RR section also reviewed Section 13.2, Training, in the FSAR and developed safety evaluation reports in this area.

1978 - 1979: I was assigned to Region II, Atlanta, Georgia and participated in a Pilot Test Program for regionalization of OLB functions. I was responsible for all licensed operator and senior operator renewals as well as changes to requalification programs in Region II. I developed and conducted examinations on all types of reactors, including the use of simulators, in the Region. Shortly after the Three Mile Island, Unit 2, accident, I was detailed as part of the NRC team at TMI for several weeks. Due to large demands on the OLB staff at Headquarters, the Pilot Test Program was suspended in the fall of 1979 and I returned to Headquarters as the PWR (Westinghouse) Section Leader. I was employed in this capacity until February of 1982.

1982 - Present: I am currently assigned as a Systems Engineer (Training and Assessment). This position requires: review of licensee's applications in Chapter 13.2 of the FSAR and preparation of Safety Evaluation Reports, review of changes to the licensee's requalification programs, response to Regional reports to provide resolution on the interpretation of training requirements. I have been recently assigned as a reviewer of Shift Advisor training programs. I have also participated in review of the ATWS event at Salem and the review of PTS training at H.B. Robinson and Calvert Cliffs. In addition, I have participated in the review of training programs at TMI.

Publications: I have contributed to several NUREGs published by the NRC.

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NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

..................................:

TESTIMONY OF WAYNE HODGES

Q. What is your name?

A. My name is Marvin Wayne Hodges

Q. What is your position at the NRC?

A. I am employed as a Section Leader in Section B of the Reactor Systems Branch in the Division of Systems Integration.

Q. What are your technical qualifications?

A. I graduated from Auburn University with a Mechanical Engineering

Degree in 1965. I received a Master of Science Degree in Mechanical Engineering from Auburn University in 1967. I am a registered professional engineer in the State of Maryland (No. 13446).

In my present work assignment at the NRC, I supervise the work of five graduate engineers. My section is responsible for the review of primary and safety systems for boiling water reactors. I have served as principal reviewer in the area of boiling water reactor systems. I have also participated in the review of analytical models used in the licensing evaluations of boiling water reactors and I have the technical review responsibility for many of the modifications and analyses being implemented on boiling water reactors post Three Mile Island Unit 2 accident.

As a member of the Bulletins and Orders Task Force, which was formed after the TMI-2 accident, I was responsible for the review of the capability of BWR systems to cope with loss of feedwater transients and small-break-loss-of-coolant accidents.

I have also served at the NRC as a reviewer in the Analysis Branch of the NRC in the area of thermal-hydraulic performance of the reactor core. I served as a consultant to the RES representative to the Program Management Group for the BWR blowdown emergency core cooling program.

Prior to joining the NRC staff in March 1974, I was employed by E.I. DuPont at the Savannah River Laboratory as a research engineer. At SRL I conducted hydraulic and heat transfer testing to support operation of the reactors at the Savannah River Plant. I also performed safety limit calculations and participated in the development of analytical models for use in transient analyses at Savannah River. My tenure at SRL was from June 1967 to March 1974.

From September 1965 to June 1967, while in graduate school, I taught courses in thermodynamics, statics, mechanical engineering measurements, computer programming, and assisted in a course in the history of engineering. During the summer of 1966, I worked at the Savannah River Laboratory doing hydraulic testing.

Q. What is the purpose of your testimony?

A. The purpose of this testimony is to describe NRC practice in applying the single failure criterion and to discuss the applicability, or lack thereof, of the single failure criterion to Suffolk County and the State of New York emergency diesel generator load contention a(iv). That part of the contention states:

"Contrary to the requirements of 10 CFR 50, Appendix A, Generic Design Criterion 17 ... Electric Power Systems, the emergency diesel generators at Shoreham ("EDGs") with a maximum "qualified load," of 3300 kW do not provide sufficient capacity to assure that the requirements of clauses (1) and (2) of the first paragraph of GDC-17 will be met in that:

(a) LILCo's proposed "qualified load" of 3300 kW is the maximum load at which the EDG may be operated, but is inadequate to handle the maximum load that may be imposed on the EDGs because:

(iv) Operators may erroneously start additional equipment;"

Q. What is meant by a single failure?

A. Single failure is defined in 10 CFR Appendix A as follows:

"A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety function. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components work properly), results in a loss of capability of the system to perform its safety functions."

Q. How is the single failure criterion used?

A. Application of the single failure criterion involves a systematic search for potential single failure points. The objective is to search for design weakness which could be overcome by increased redundancy or use of alternate systems. The single failure criterion is used to ensure the reliability of those systems which are essential to the safety of the plant.

Q. Are operator errors included in the single failure analysis?

A. No, operator errors are not included in the single failure analysis. Single failures are postulated to occur only in components, consistent with the definition of single failures in Appendix A to 10 CFR 50.

Q. How are operator errors accounted for in the design of the plant?

A. Operator errors are accounted for in the design of the plant in a number of ways. First, for actions that must be accomplished on a relatively short time scale and are necessary to mitigate transients and accidents, the staff policy has been to eliminate the need for operator action by automating the action. By not challenging the operator with an action on a relatively short time frame, the potential for operator error is greatly reduced so it is not considered in the context of the design. Second, for situations in which operator actions are relied upon for event mitigation, the staff ensures that procedures and guidelines provide the necessary guidance to the operator to take the correct actions, and that the operators have been properly trained in the action. Third, in the event the staff determines that reasonable assurance does not exist that an operator would not make an error, then the staff would require that (1) the postulated operator error be considered in the design, (2) the design be modified in order to acceptably accommodate the postulated operator error, (3) that procedures and training be instituted such that the potential for operator error is reduced to a acceptable level, or (4) that assurance be provided that the operator could take the necessary corrective actions to remedy the original error in a reasonable time frame without unacceptable consequences resulting. Finally, a spectrum of operator errors are inherently considered as part of the single failure assumption. That is, because the staff does not require the cause of single failures to be specified, it is obvious that many single failures could be considered to be caused by operator error as well as other causes.

Q. Are operator errors considered in addition to another failure in a single failure analysis?

A. No. The purpose of the single failure analysis is to gain greater assurance of system reliability through redundancy. Operator reliability would not be assured by such an analysis. Operator reliability depends first on having well designed equipment. Then good procedures and training will assure operator reliability. The systems analysis must assume that good procedures exist for the operator to follow and that the operator is trained on those procedures.

Q. Are cognitive operator errors considered in single failure analyses?

A. Not directly. As stated before, the purpose of the single failure analysis is to assure system reliability through redundancy. Cognitive operator errors must be addressed through training and procedures. The operator must understand the system well enough to understand the effects of actions he/she is taking and to recognize symptoms which indicate problems; he/she must also have good procedures to aid in carrying out his/her mission.

Q. Does Suffolk County and New York State emergency diesel generator load contention a(iv) raise an impermissible challenge to the single failure criterion?

A. No. The single failure criterion is not applicable to the treatment of operator errors. Operator errors would normally be considered in the design of a system so that the system is tolerant of operator errors through either procedures or design or both. An example of this is the design of low pressure systems which interface with high pressure systems. Interlocks are provided to prevent opening valves between the systems when the pressure in the high pressure system is above the design pressure of the low pressure system. The interlocks are generally single failure proof and will protect against many operator errors as well as system failures. However, the systems may still be susceptible to common mode maintenance errors. Proper training and procedures are needed to protect against such errors.

Q. Are there interlocks or permissives which prevent operators from loading the emergency diesels at Shoreham to more than 3300 kW?

A. No.

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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)	)	

JOINT TESTIMONY
of
SPENCER H. BUSH, ADAM J. HENRIKSEN, AND PROFESSOR ARTHUR SARSTEN
on
LOAD CONTENTIONS CONCERNING TDI EMERGENCY DIESEL GENERATORS
at the
SHOREHAM NUCLEAR POWER STATION

CONTENTS

INTRODUCTION OF WITNESSES

Q. Please state your names, your business addresses, and your professional qualifications.

A. (Bush) My name is Spencer H. Bush. I am self-employed, under the firm name of Review and Synthesis Associates, Richland, Washington. A summary of my professional qualifications and experience was submitted as Attachment 2 to Volume 1 of the joint testimony filed by the NRC staff in August 1984.

A. (Henriksen) My name is Adam J. Henriksen. I am self-employed, under the firm name of Adam J. Henriksen, Inc., Fox Point, Wisconsin. A summary of my professional qualifications and experience was submitted as Attachment 3 of the joint testimony referenced above.

A. (Sarsten) My name is Arthur Sarsten. I am a Professor of Internal Combustion Engines at the Norwegian Institute of Technology, Trondheim, Norway. A summary of my professional qualifications and experience was submitted as Attachment 5 of the joint testimony referenced above.

SCOPE OF TESTIMONY

Q. What is the scope of your testimony?

A. (All) Our testimony addresses the following parts of Suffolk County's load contention as admitted by the Atomic Safety and Licensing Board:

> Contrary to the requirements of 10 C.F.R. Part 50, Appendix A, General Design Criterion 17 -- Electric Power Systems, the emergency diesel generators at Shoreham ("EDGs") with a maximum "qualified" load of 3300 kW do not provide sufficient capacity and capability to assure that the requirements of clauses (1) and (2) of the first paragraph of GDC 17 will be met, in that
>
> (a) LILCO's proposed "qualified load" of 3300 kW is the maximum load at which the EDGs may be operated, but is inadequate to handle the maximum load that may be imposed on the EDGs because:
>
> (i) intermittent and cyclic loads are excluded;
>
> (ii) diesel load meter instrument error was not considered;
>
> (iii) operators are permitted to maintain diesel load at 3300 kW ±100 kW; and
>
> (iv) operators may erroneously start additional equipment.
>
> (c) The EDG qualification test run performed by LILCO was inadequate to assure that EDGs are capable of reliable operation at 3300 kW because:
>
> (i) DG 103 block was not subjected to the entire 740 hours of testing;
>
> (ii) the test results on the DG 103 block are not transferable to the DG 101 and 102 blocks;
>
> (iii) operators were permitted to control the diesel generators at 3300 kW ±100 kW during the test; and
>
> (iv) instrument accuracy was not considered.

SUMMARY OF TESTIMONY

Q. Please summarize your testimony on these contentions.

A. (All) Our summary testimony is provided under the two subheadings that follow.

FATIGUE LIFE OF CRANKSHAFTS IN THE SHOREHAM EDGs

From our review of LILCO's testimony and data logs, we believe that EDG 103 was, in fact, operated at a nominal, instrument-indicated load of 3300 kW during that portion of the 1×10^7-cycle confirmatory test claimed by LILCO to have been conducted at the 3300-kW load level. We understand that the wattmeter may oscillate approximately ±100 kW around the value at which the load is set, presumably because this is as close as the load can be controlled without blocking the governor. Based on wattmeter calibration data, the actual load could have differed from the indicated load by about ±70 kW. In the context of the overall test loads included in the 10^7 cycles and the order in which they occurred, however, we view these deviations from 3300 kW as of no consequence.

In our opinion, EDGs 101, 102, and 103 are suitable for nuclear standby service at the "qualified" load of 3300 kW. This opinion is subject to the surveillance and maintenance recommendations documented in the following technical evaluation report, which we assisted in preparing: <u>Review and Evaluation of Transamerica Delaval, Inc., Diesel Engine Reliability and Operability - Shoreham Nuclear Power Station Unit 1</u>, PNL-5342, dated December 1984. As noted on pages 4.24 through 4.25 of that report, "...the replacement crankshafts for

EDG 101, EDG 102, and EDG 103 are acceptable for their intended service, provided that they are not operated during engine tests at loads in excess of the qualified load of 3300 kW." We believe that this restriction is necessary to avoid routine operation of the crankshafts at loads in excess of the load at which one crankshaft has been successfully tested.

Accordingly, we recommend that the permissible load for engine tests, including surveillance tests at the qualified load, be no higher than 3300 kW as read on control room instrumentation. We understand that the wattmeter may oscillate approximately ±100 kilowatts around the value at which the load is set, as discussed above. In our opinion these oscillations during routine tests will not be detrimental to engine reliability, provided that the indicated mean load is no higher than 3300 kW.

Loads at which EDG 103 was operated as part of the confirmatory test to 1×10^7 cycles, and the post-test examination that revealed no evidence of damage to the crankshaft or other key engine components, provide a basis for drawing conclusions about the capability of the EDGs for emergency operation at loads above the qualified load. EDG 103 sustained over 220 hours (approximately 3×10^6 cycles) at instrument-indicated loads of 3500 kW and above. With a conservative application of instrument error from calibrations performed by LILCO preceding and following the time the higher-load testing was performed, we estimate that the actual load during this period was at least 3430 kW. If cracks had initiated during this testing, it is likely that they would have propagated during subsequent operation at approximately 3300 kW for the time necessary to bring the total cycles to 1×10^7. But no cracks were found in the post-test inspection of the crankshaft.

In light of these results, and taking into consideration the small but inevitable differences in the properties of the three crankshafts, it is our opinion that it would be within the demonstrated capability of the engines to operate at loads to 3430 kW for an hour or so if the engines were needed to carry such loads under emergency conditions. This comment does not apply for routine operation of the engines, including engine testing, for which we recommend a load limit of 3300 kW as discussed earlier in this summary.

The testing performed on EDG 103 does not provide an adequate basis for drawing conclusions about the effects on the EDGs of loads higher than 3430 kW. However, an additional observation may be made based on other considerations. It is generally accepted in the technical literature on fatigue and cumulative damage in metals that momentary overloads, even those approaching the ultimate tensile strength of the metal, can be sustained without failure. This literature provides a basis for confidence that brief excursions (less than 1 minute) of the Shoreham engines to loads as high as 3900 kW under emergency conditions would not compromise engine operability.

If an engine were operated at high overload for a longer period during an emergency, its capability to meet the load profile throughout the emergency would depend on whether or not a crack would initiate in the crankshaft during the overload and propagate to failure before the engine was no longer needed. The available information does not provide a basis for us to comment with confidence on this scenario. However, overloads to 3900 kW for up to 1 hour under emergency conditions followed by much lower loads in accordance with LILCO's predicted LOOP/LOCA profile are believed to be sustainable. Any crankshaft that is subjected to more than a momentary overload approaching this level should receive a thorough nondestructive examination before it is returned to service.

CYLINDER BLOCKS

The replacement EDG 103 block was not subjected to the entire qualification test performed on the EDG 103 engine. Nevertheless, the absence of any reportable indications in the block top after more than 500 hours of operation at or above 3300 kW provides significant evidence that the replacement block is suitable for service at the qualified load. If further operation beyond the most recent inspection does not exceed the FaAA-recommended inspection interval before the end of the first fuel cycle, the top of the replacement block need not be reinspected until the first shutdown for refueling. It is also unnecessary, in our opinion, to monitor cam gallery cracks in the replacement block. The known cam gallery cracks in this block have not been repair-welded, and, therefore, residual stress fields that may be associated with repair welds have not been introduced into the block material.

The replacement EDG 103 block was more suitable than either the EDG 101 block or the EDG 102 block for the tests that LILCO conducted to obtain data on compressive and alternating stresses in the camshaft gallery. Use of either of the latter two blocks for the cam gallery tests would have involved the installation of strain gages over repair welds rather than over base metal. However, the test of EDG 103 at qualified load did not contribute to resolution of questions concerning the ligament cracks in the top surfaces of the EDG 101 and 102 blocks, the potential for developing stud-to-stud or stud-to-end cracks in those blocks, or the circumferential cracks reported in the original EDG 103 block.

Our conclusions expressed previously in written testimony regarding the EDG 101 and 102 blocks remain unchanged. In our opinion, the 101 and 102 blocks are adequate for service subject to certain caveats on surveillance of known cracks. Following any period of operation of EDG 101 or EDG 102 at or above 50% of qualified load, visual (with the naked eye) and eddy-current inspections should be performed on those portions of the block top that are accessible between cylinder heads. The purpose of these inspections is to verify the continued absence of detectable cracks between studs of adjacent cylinders. In addition, the behavior of several representative cracks in the camshaft galleries of the EDG 101 and 102 blocks should be monitored. If no changes indicative of crack growth are observed over the first fuel cycle, the need for continued monitoring of the cam gallery cracks should be reconsidered by the NRC staff.

Our opinion expressed in previous testimony is also unchanged regarding circumferential cracks of the type found in a cylinder liner counterbore of the original EDG 103 block. If such cracks were to develop in any of the three blocks currently in service, it is highly unlikely that they would represent a hazard to EDG reliability. They would be expected to propagate only a short distance into a region of compressive stress and stop. At any time a liner is removed from any of the three engines, however, it would be prudent to perform an appropriate nondestructive examination of the landing of the block. If a circumferential indication is found, an attempt should be made to characterize the depth and length of the indication through appropriate nondestructive tests. However, we do not advocate removal of cylinder liners for the sole purpose of this inspection.

TESTIMONY ON CONTENTIONS

Q1. How is your testimony organized?

A1. (All) The testimony is presented in two general parts concerning 1) the crankshaft and 2) the cylinder block.

I - CRANKSHAFT

Q2. What issues are addressed in this part of your testimony?

A2. (All) This part of the testimony deals with 1) conclusions that may be drawn from the qualification tests, and 2) the fatigue life of the crankshafts currently installed in the Shoreham TDI diesel engines, designated as EDGs 101, 102, and 103. Item 1 is relevant to the contentions (c)(i) through (iv) and Item 2 is relevant to contentions (a)(i) through (iv).

Conclusions that May be Drawn From Confirmatory Testing

Q3. Can you comment on the purpose of the confirmatory tests done by LILCO to accumulate 10^7 operating cycles on EDG 103?

A3. (All) It is our understanding that these tests were conducted by LILCO primarily to provide unequivocal evidence that the high-cycle fatigue endurance limit of the crankshaft used in EDGs 101, 102, and 103 is at or above 3300 kW. The tests also included strain gage measurements to determine if the stress field in the cam gallery region of the block is compressive. These cam gallery tests are discussed in a later section of this testimony.

Q4. Have you reviewed the procedures and results pertaining to the confirmatory tests done by LILCO to accumulate 10^7 operating cycles on EDG 103?

A4. (All) Yes. Our review of the test results has been provided to the Board in two reports, namely Post-Test Examination of Transamerica Delaval, Inc. Emergency Diesel Generator 103 at Shoreham Nuclear Power Station for U.S. Nuclear Regulatory Commission Staff, by A. J. Henriksen, B. J. Kirkwood, W. W. Laity, P. J. Louzecky, J. F. Nesbitt, and L. G. Van Fleet, dated December 3, 1984, and Post-Test Examination of the Transamerica Delaval, Inc. Emergency Diesel Generator 103 Piston Skirts and Related Components at Shoreham Nuclear Power Station for U.S. Nuclear Regulatory Commission Staff, by A. J. Henriksen, B. J. Kirkwood, W. W. Laity, P. J. Louzecky, J. F. Nesbitt, and L. G. Van Fleet, dated December 14, 1984. Our review of the procedures is based on LILCO's letter to NRC (Harold Denton) dated October 18, 1984, concerning the confirmatory test, and information provided in test data sheets and supporting procedures regarding the calibration of electrical switchboard instruments.

Q5. Why was it not possible to draw conclusions regarding the acceptability of the crankshafts from calculations alone?

A5. (Sarsten) Crankshaft calculations involve uncertainties arising from the complex geometry of crankshafts and the variations in torque, bending loads, and other relevant input data. A large factor of safety must be employed to accommodate these uncertainties. It appears to me that the analytical evidence alone does not provide a sufficient basis for concluding that the crankshafts are adequate for the qualified load of 3300 kW. An unequivocal answer can be supplied only by an engine test for a sufficient time to accumulate 10^7 operating cycles.

Q6. Regarding the tests conducted by LILCO at a nominal 3300 kW, do you believe that they can be proven to have been at that value?

A6. (All) No. We noted several points that could affect the certainty of the tested value:

1. There was uncertainty with respect to whether operators had the flexibility during the confirmatory tests to operate at 3300 ± 100 kW.

2. Instrument uncertainties could have introduced an error of up to 2.5% of full-scale power readings.

3. LILCO reported that 20 hours were run at loads in the range of 3250 to 3300 kW and that 81 hours were run at loads between 3300 and 3400 kW.

Q7. Have you resolved these questions?

A7. (Henriksen) I believe so. The points just identified have been addressed. First, based on a review of the testimony and the data logs provided, I believe LILCO operators did operate most of the time with the wattmeter indicating a load of 3300 kW. This is based on my belief that the flexibility provided by NRC in conducting surveillance tests at 3300 kW ±100 kW does not really mean that the load will be set at 100 kW above or below 3300 kW during that test. Rather, as I understand it, when set at 3300 kW, due to the mode of operation described in LILCO's testimony, the wattmeter oscillates between 3200 and 3400 kW. This is probably as close as the load can be controlled unless the governor load limit is blocked.

I have also reviewed the level of possible errors involved in the load measuring system. According to LILCO's testimony, the wattmeter instrument error could be as much as ±2% of full-scale or ±112 kW. An additional error of ±0.5% or ±28 kW in the remainder of the instrument loop could result in a total of ±2.5% or ±140 kW error in measuring the load. However, the calibration data furnished for the wattmeter, dated November 10, 1983, October 1, 1984, and January 4, 1985, indicated that the error in the meter never exceeded 40 kW in the 3000 to 4000 kW load range. Thus, including the possible 28 kW error in the remainder of the loop, the total instrument error appears to not have exceeded ±1.25% or ±70 kW during any period of operation of this particular engine since November 10, 1983.

The 20 hours of operation reported to be below 3300 kW is considered to be sufficiently few that they are of little or no significance to the question of the tested load, especially since there were 81 hours of operation above 3300 kW.

Q8. Does the possibility that due to instrument errors the confirmation test may have been conducted at a load as low as 3230 kW mean that the endurance limits for the crankshafts cannot be confirmed to meet or exceed 3300 kW?

A8. (Bush) No. I believe the crankshaft is qualified for its intended service even though some of the confirmatory test data may have been accumulated at loads slightly below 3300 kW. As I will testify in a later section, I am convinced from my analysis of engine load data that EDG 103 has operated at or above an instrumented-indicated load of 3500 kW for about 3 x 10^6 cycles with no evidence of damage to the crankshaft. This strongly suggests that the endurance limit is at or above 3430 kW, accounting for instrument error.

Additional testing of 7×10^6 cycles at engine loads near 3300 kW would have been sufficient to propagate any cracks that may have been present because the crankshaft stresses at 3300 kW are quite close to those at 3500 kW. Therefore, I do not consider it significant that some of the confirmatory testing may have occurred at loads somewhat below 3300 kW.

Fatigue Life of Crankshafts in the Shoreham EDGs

Q9. Have you reviewed the testimony of the County and LILCO regarding the load profiles that the Shoreham EDGs will be required to provide?

A9. (Bush, Sarsten, Henriksen) Yes. Generally we understand the engines may be subjected to loads in the following categories:

1. Load spikes equivalent to 3900 kW due to sequenced starting of large cooling pumps for the first 30 to 60 seconds of a LOOP/LOCA event.

2. Short time intermittent and cyclic loads for a few minutes that may exceed by a few percent the "qualified load", taken here as 3300 kW.

3. LOOP/LOCA loads, assumed to be at or below 3300 kW after the first few minutes.

4. Loads that may result from operator error during the first hour of a LOOP/LOCA event, taken as 3800 to 3900 kW for times of 40 to 60 minutes.

5. Periodic testing loads of 3300 kW to meet NRC Regulations.

Q10. Do you believe the engines (EDGs 101, 102, and 103) can sustain loads of Category 1 as described above?

A10. (Bush) Short-term loads as high as 3900 kW for less than a minute under emergency conditions are not considered to be a problem. Almost all texts related to fatigue and to cumulative damage in metals cite the effects of momentary overloads. An example is Collins Failure of Materials in Mechanical Design (1981). Figure 1, taken from Collins (1981, p. 293, Figure 8.27), illustrates the prestressing effect of momentary overloads on existing cracks and their subsequent delay in propagation.

Short-term high loads, even those approaching the ultimate tensile strength, do not generally produce cracks and may, in fact, provide a plastic zone around any existing crack that retards its growth. The preceding condition markedly exceeds the short-term achievable overloads of these EDGs. It is my conclusion, therefore, that loads such as those identified in Category 1 are not of concern.

Q11. Do you believe the Shoreham TDI EDG crankshafts can sustain loads identified in Category 2 as described above?

A11. (Bush) I would like to offer some background information rior to answering this question. I have carefully reviewed the operating history of the Shoreham EDGs, particularly noting the operating time at engine loads at and above 3500 kW. In the case of EDG 103, which has undergone extensive post-test examination showing no damage to the engine (particularly the crankshaft), I note that the engine has sustained over 3×10^6 cycles at loads at or exceeding 3430 kW when conservative assumptions regarding instrument error are included as discussed earlier.

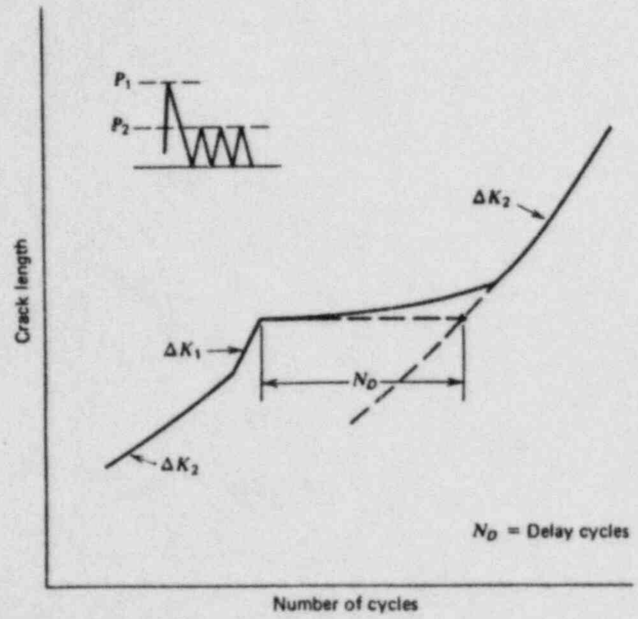

FIGURE 1. Delay in Crack Growth Following the Application of Single Overload

Source: J. A. Collins, Failure of Materials in Mechanical Design - Analysis, Prediction, Prevention, 1981, p. 293, Figure 8.27.

The loads and corresponding hours at which EDG 103 is reported to have operated are as follows:[a]

Load	Hours
Approximate hours at 3500 kW	119
Approximate hours at loads greater than 3500 kW	101
Approximate hours at 3900 kW	7

Any of several approaches may be used to predict cumulative fatigue damage from these loads. Miner's rule, more correctly termed the Palmgren-Miner cyclic-ratio summation theory, has been used for many years to predict the fatigue (endurance) limit of materials. An alternative method that provides better correlation with experimental data is the Manson approach, which takes into account the loading sequence. The predicted fatigue limit using the latter approach for the EDG 103 crankshaft would vary markedly depending on the sequence of application of the loads noted in the preceding summary. We are unaware from available information what the actual sequence was.

A conservative view is to assume that the beginning of the high-cycle fatigue limit is less than 3×10^6 cycles, and to define the lower bound of the fatigue limit as that associated with the lowest load at which EDG 103 was operated during the first 3×10^6 cycles. This would set the lower-bound value from the EDG 103 test at 3430 kW, based on an assumed instrument error of ±70 kW applied to the indicated load of 3500 kW.

(a) Pacific Northwest Laboratory, Review and Evaluation of Transamerica Delaval, Inc., Diesel Engine Reliability and Operability - Shoreham Nuclear Power Station Unit 1, PNL-5342, December 1984 (p 4.22).

Table 1 is a summary of data from six references on the high-cycle fatigue limit for several ferrite steels. A significant message from this data is that the onset of the fatigue limit is close to 1×10^6 cycles, regardless of the ferritic alloy, heat treatment, or surface hardening treatment. Note that several of the values are for aircraft or automobile crankshafts.

As illustrated in Figure 2, the fatigue limit of ferrite steels is essentially constant as a function of the number of cycles above the onset of high-cycle fatigue. This is unlike nonferrous metals, which have no clearly defined fatigue limit with time.

The steel used in the EDG 103 crankshaft is ABS Grade 4S, which corresponds roughly to an AISI-5050 steel in composition. The tensile strength is about 100 ksi and the yield strength about 60 ksi. The mechanical properties would correspond to some of the 4000 series steels cited in Table 1, and, therefore, one would anticipate similar initiation of the fatigue limit near 1×10^6 cycles.

LILCO's nondestructive examinations of the EDG 103 crankshaft following the 10^7-cycle test provide evidence that cracks had not initiated in the crankshaft during the initial 3×10^6 cycles at loads at or above 3500 kW as read on the wattmeter. Because crankshaft stresses at 3500 kW are not substantially different from stresses at 3300 kW (as discussed in response to Question 12), subsequent operation at the latter load to bring the total cycles to 10^7 would have been sufficient to cause propagation of cracks formed at the higher load. This is further confirmation that the high-cycle fatigue limit is at or above the value corresponding to 3500 kW minus known instrument error, or 3430 kW.

TABLE 1. Location of the Initiation of High-Cycle Fatigue (Endurance) Limit for Several Ferrite Steels

Reference	Beginning of Fatigue Limit x 10^6 Cycles	Material	Comments
(1)	1.0	1047 Steel	
(2)	~3.0	4340	Vacuum melted - longitudinal specimens
	~3.0	4340	Vacuum melted - transverse specimens
	~0.9	4340	Air melted - longitudinal specimens
(3)	~1.5	4340	Completely reverse S-N curve
(4)	~0.3	3130	Temper embrittled
	~0.8	3130	Non-temper embrittled
(5)	2.0	0.78% C	Spheroidized
	2.5	0.78% C	Pearlitic
(5)	1.5	4140	Quenched and tempered
	2.0	4140	Shotpeened
	2.5	4140	Nitrided
(5)	0.7	(4140,x4340, VCM)[a]	Quenched and tempered
	1.0	(4140,x4340, VCM)[a]	Shot-peened
	1.5	(4140,x4340, VCM)[a]	Nitrided, polished nitrided
	~3.0	(4140,x4340, VCM)[a]	Nitrided
(5)	0.8	4340	Automobile crankshaft - normal heat treatment
	0.7	4340	Automobile crankshaft - shot-peened
	~2.0	4340	Automobile crankshaft - nitrided
(5)	1.5	4340	Transverse specimens from crankshaft
	0.2	1.20% C	Quenched and tempered

(a) Above are torsional fatigue results on aircraft engine crankshafts including 4140 series.

TABLE 1. (contd)

Reference	Beginning of Fatigue Limit x 10^6 Cycles	Material	Comments
(6)	0.9	3420	Quenched and tempered
	1.0	1050	Quenched and tempered
	1.0	4130	Normalized
	1.5	Structural steel	-
	1.5	Alloy struc. steel	-
	~2.0	Cast iron	-

(1) Hayden, H. W., et al. 1965. "Mechanical Behavior". Volume III in The Structure and Properties of Materials. John Wiley & Sons, New York, New York.

(2) Reed-Hill, R. F. 1964. Physical Metallurgy Principles. Van Nostrand, New York, New York.

(3) Collins, J. A. 1981. Failure of Materials in Mechanical Design - Analysis Prediction, Prevention. John Wiley & Sons, New York, New York.

(4) Hollomon, J. H., and L. D. Jaffee. 1974. Ferrous Metallurgical Design. John Wiley & Sons, New York, New York.

(5) American Society of Metals. 1961. "Properties and Selection of Metals". Volume 1 in ASM Metals Handbook. Novelty, Ohio.

(6) Marks, L. S. 1941. Mechanical Engineers' Handbook. 4th ed. McGraw-Hill, New York, New York.

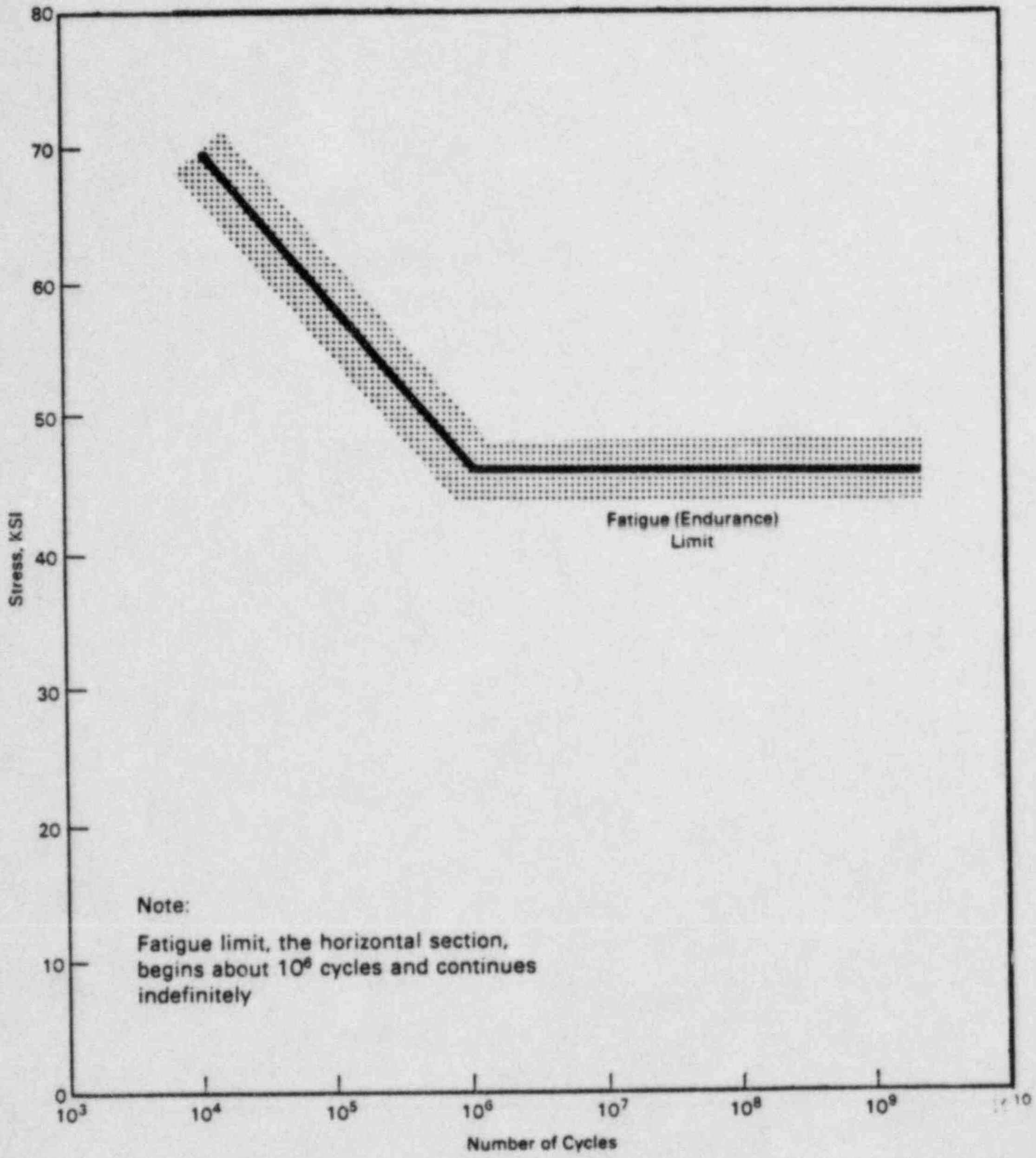

FIGURE 2. Typical High-Cycle Fatigue Curve for a Ferritic Steel (1050 AISI)

The point of the background discussion is now clear. In my opinion, the Category 2 engine loads that may result from intermittent and cyclic demands in the vicinity of 3350 to 3400 kW for times up to one hour or so are below the probable high-cycle fatigue limit. Therefore, loads in Category 2 are not of concern.

Q12. Can you quantify the relative stresses at 3300 kW and 3500 kW?

A12. (Sarsten) If one takes the bending stresses as employed and interpreted by Det Norske Veritas for the Shoreham crankshafts in their report 84-0099A of September 17, 1984, and the maximum firing pressures as read from TDI test curves dated March 19, 1976, for a Shoreham engine, then the relative calculated bending stresses are 20,450 psi and 21,120 psi for 3300 kW and 3500 kW, respectively.

Q13. Do you believe the EDGs can sustain the loads identified in Category 3 above?

A13. (Bush) As defined in the response to Question 9, all loads in Category 3 are at or below 3300 kW. I believe the endurance limit for these crankshafts is above this value. Hence, the Category 3 loads are not of concern.

Q14. The engine loads that may result from operator error (e.g., Category 4) could exceed the high-cycle fatigue limit. Do you believe the crankshafts will sustain these loads for periods up to an hour and still have the ability to meet the succeeding load challenge of a LOOP/LOCA?

A14. (Bush) I believe the crankshaft can survive up to an hour of overload to about 3900 kW without crack initiation, but the probability of crack initiation cannot be quantified. It is a function of parameters such as previous load history and metallurgical properties. The question then is, if a crack initiates during a LOOP/LOCA, will it propagate to the point of engine shutdown before the engine is no longer needed? My engineering judgment is that the combination of a Category 4 transient operation followed by time at lower load/time profiles such as the LOOP/LOCA demand profile should not lead to crankshaft failure. The only way to quantify this judgment would be to conduct a three-dimensional finite element analysis combining the LOOP or LOOP/LOCA load histories that were imposed on a crankshaft having an initial crack and determine the final crack size.

I feel that any crankshaft that is subjected to a sustained overload approaching Category 4 should be given careful surface and volumetric non-destructive examination prior to returning it to service.

Q15. What LOOP/LOCA load profile did you consider in evaluating the ability of the crankshaft to sustain the assumed operator error load?

A15. (Bush) I assumed the following LOOP/LOCA load profile based on data provided in LILCO's testimony dated January 15, 1985, and the Shoreham Final Safety Analysis Report (FSAR), Tables 8.3.1-1A and 8.3.1-2:

Time	Load (kW)
Less than 1 minute	3900
1 minute to 3 minutes	3331
3 minute to 12 minutes	3266
12 minutes to 30 minutes	3265
30 minutes to 60 minutes	3253
Longer than 60 minutes	2617

Q16. Do you believe the Shoreham EDGs can sustain the NRC required monthly and refueling-outage testing at the qualified load of 3300 kW, identified in the response to Question 3 as Category 5 loads?

A16. (Bush, Sarsten, Henriksen) Yes. These Category 5 testing loads are considered to be below the endurance fatigue limit for these crankshafts. As stated earlier, this limit is believed to be at or above 3430 kW, based on the results of the testing up through the first 3×10^6 cycles, and is certainly confirmed to be at or above 3300 kW, based on the confirmatory tests that brought the total testing cycles to over 1×10^7. Detailed comments regarding these confirmatory tests, including our views on the uncertainties with wattmeter readings, are provided earlier in this testimony.

In view of the fact that the endurance limit can be established with certainty as being only at or above 3300 kW, we feel that it would be prudent to limit surveillance testing to this value. The reason for this is that surveillance tests can add over 3×10^7 cycles during the assumed 40-year life of the Shoreham Nuclear Power Station.

II - CYLINDER BLOCKS

Q17. What is the purpose of this testimony?

A17. (Bush) This testimony addresses parts c(i) and c(ii) of the contention concerning testing of the EDG 103 block, and also addresses metallurgical considerations related to my conclusion that existing cracks in the cam gallery region of the EDG 101 and 102 blocks should be monitored.

Q18. Have you reviewed the testimonies filed by the County and by LILCO concerning the test involving the EDG 103 block, the suitability of the cylinder blocks in EDGs 101 and 102 for service at 3300 kW, and whether there is a need to monitor the cam gallery cracks in the EDG 101 and 102 blocks?

A18. (Bush) Yes.

Q19. Please summarize your conclusions on these issues.

A19. (Bush) My conclusions are as follows:

First, as I have stated previously in written testimony (filed on October 12, 1984), the replacement EDG 103 block was more suitable than either the EDG 101 block or the EDG 102 block for the tests that LILCO conducted to obtain data on compressive and alternating stresses in the camshaft gallery. Use of either of the latter two blocks for the cam gallery tests would have involved the installation of strain gages over repair welds rather than over base metal. However, the selection of EDG 103 for the test at qualified load did not contribute to resolution of questions concerning the ligament cracks in the top surfaces of the EDG 101 and 102 blocks, the potential for developing stud-to-stud or stud-to-end cracks in those blocks, or the circumferential cracks reported in the original EDG 103 block.

Second, operation of the replacement EDG 103 block for more than 500 hours at or above 3300 kW based on the meter reading, followed by LILCO's nondestructive examinations that revealed no reportable indications in the block top, provides significant evidence that the replacement block is suitable for service at the qualified load of 3300 kW. Based on the known performance of the block through the qualification test, I concur with the conclusion of

Dr. Rau and Dr. Wachob[a] that it would be appropriate to reinspect the replacement block top at intervals determined through FaAA's cumulative damage analysis.[b] This means that if further operation beyond the most recent inspection does not exceed the FaAA-recommended interval before the end of the first fuel cycle, the top of the replacement block will not have to be reinspected until the first shutdown for refueling.

Third, the conclusions I expressed in previous written testimony regarding the EDG 101 and 102 blocks are not affected by the qualification test performed with EDG 103. As I previously testified, I believe that the 101 and 102 blocks are adequate for service subject to certain caveats on surveillance of known cracks. Following any period of operation of EDG 101 or EDG 102 at or above 50% of qualified load, visual and eddy current inspections should be performed on those portions of the block top that are accessible between cylinder heads. The purpose of these inspections is to verify the continued absence of detectable cracks between studs of adjacent cylinders. In addition, the behavior of several representative cracks in the camshaft galleries of the EDG 101 and 102 blocks should be monitored. If no changes indicative of crack growth are observed over the first fuel cycle, the need for continued monitoring of the cam gallery cracks could be reconsidered by the NRC.

Fourth, I have previously expressed the opinion based on engineering judgment that circumferential cracks of the type found in a cylinder liner

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(a) Additional Cylinder Block Testimony of Dr. Duane P. Johnson, Dr. Charles A. Rau, Jr., Milford H. Schuster, Dr. Harry F. Wachob and Edward J. Youngling on Behalf of Long Island Lighting Company, January 15, 1985, at 10.

(b) This analysis is presented in the FaAA report Design Review of TDI R-4 and RV-4 Series Emergency Diesel Generator Cylinder Blocks, the most recent revision of which is FaAA-84-9-11.1 dated December 1984.

counterbore of the original EDG 103 block do not represent a hazard to EDG reliability. My opinion on that issue remains unchanged. Similar cracks may also occur in the EDG 101 and 102 blocks because of the high stress concentration associated with the geometry of the cylinder liner landing. They may occur even in the replacement EDG 103 block, although the stress concentration in the replacement block appears to be less severe. At any time a liner is removed from any of the three engines, it would be prudent to perform an appropriate nondestructive examination of the landing in the block. If a circumferential indication is found, an attempt should be made to characterize the depth and length through appropriate nondestructive tests. However, I do not advocate removal of cylinder liners for the sole purpose of this inspection.

Monitoring of Cam Gallery Cracks in EDGs 101 and 102

Q20. How is your testimony organized on this topic?

A20. (Bush) I first will comment on the examination[a] performed by Walter C. McCrone Associates, Inc. of a cam gallery crack specimen removed from the original EDG 103 block. I will next briefly summarize my assumptions and conclusions regarding the origin and characteristics of the cam gallery cracks. Finally, I will present my conclusions regarding the need for monitoring cam gallery cracks in the blocks of EDGs 101 and 102, and my reasons for those conclusions.

(a) Walter C. McCrone Associates, Inc., Cast Iron Analysis re LILCO vs Suffolk Company (sic), MA number 13747, dated January 11, 1985.

Comments on Testing Performed by Walter C. McCrone Associates, Inc.

The test results reported by McCrone provide unequivocal evidence that the predominant oxide in the samples removed from the crack surface was magnetite. The x-ray diffraction patterns are unambiguous and can be readily interpreted by an analyst who is trained in the field of x-ray diffraction. The McCrone laboratories are well known at the Pacific Northwest Laboratory as having competence in conducting quantitative iron-oxide measurements of the type requested by the County.

Assumptions and Conclusions Regarding Origin and Characteristics of Cam Gallery Cracks

Based on the above-mentioned test results, I have concluded that the crack examined in the sample removed from the original EDG 103 cylinder block was formed during cooling of the casting. There was no evidence of an oxide film formed at low temperatures, which could have been indicative of crack propagation after the block was placed in service. The absence of the latter oxide film tends to confirm that the crack is in a compressive stress field as determined analytically and experimentally by FaAA.

Because the original EDG 103 block exhibited degraded metallurgical properties as confirmed by the morphology of the Widmanstaetten structure, it is reasonable to assume the following:

1. The tensile properties of the typical Grade-40 cast iron in the EDG 101 and 102 blocks are superior to those of the degraded Grade-40 cast iron in the original EDG 103 block. The Grade-45 cast iron in

the replacement EDG 103 block compares even more favorably in this regard. If one reasonably assumes that the hot tensile properties of the EDG 101, 102, and replacement 103 blocks would also be better than those of the original EDG 103 block, the depth of cam gallery cracks in the former would be expected to be shallower than those in the latter.

2. With the evidence that cam gallery cracks in the original EDG 103 block are hot tears that did not propagate, and recognizing the superior materials properties of the EDG 101, 102, and replacement 103 blocks, it is reasonable to assume that the cracks in the latter blocks are also hot tears and that these cracks have not grown in service.

Conclusions Regarding the Need for Monitoring Cam Gallery Cracks

Based on the information summarized above, I conclude that the existing cam gallery cracks in the EDG 101, 102, and 103 cylinder blocks would not be expected to grow under normal operating conditions. Nevertheless, I believe that monitoring of the cam gallery cracks in EDGs 101 and 102 is necessary for the reasons listed below. I do not believe it is necessary to monitor cam gallery cracks in EDG 103, because the known cracks in the replacement block have not been repair-welded.

1. The inferences and conclusions regarding crack behavior are based on detailed examination of one crack in the original EDG 103 block. This is insufficient data on which to draw conclusions with certainty regarding the other EDG blocks.

2. Associated with the known repair welds in the cam galleries of the EDG 101 and 102 blocks are residual stress fields of an undetermined nature. These stress fields could influence crack propagation.

3. Cracks in the cam gallery represent a degraded condition. In my opinion the known data on these cracks where weld repairs have been made is insufficient to establish what will or will not happen to these cracks over time. My concern is related to the possibility of an initial lengthening of the cracks into stress fields of decreasing compression or, possibly, tension.

4. Certain postulated crack growth patterns ultimately could lead to a loss of function of a diesel generator. I recognize this is improbable, particularly when coupled to the low probability of a LOOP/LOCA. However, crack monitoring will provide confirmation as to whether or not the cracks continue to be benign. The action needed to perform the monitoring is straightforward, and I believe that it would be consistent with good practice for safety-related equipment in nuclear service.

In my opinion, the preferred approach for monitoring the cracks would be to install crack-opening displacement gages at the weld overlays on the second camshaft bearing saddle inboard of each end of the engine. These saddles are representative, and they are much more accessible than saddles toward the middle of the engine for any servicing of gages that may be required. The gages should be monitored during monthly engine tests.

Other methods of monitoring may also be acceptable. One alternative approach would be to monitor the depth of representative cracks (e.g., at locations described above) with an appropriate surface probe (e.g., a TSI depth gage), and also monitor crack length (parallel to the longitudinal axis of the engine) using magnetic particle or liquid penetrant examinations. Depth measurements taken in this manner may lack accuracy, but the combination of depth measurements and length measurements would probably be sufficient to show any significant changes in crack size. To obtain the desired information in this manner with minimal disruption of engine availability (due to the need to remove access covers), it would be sufficient to take these measurements every 3 months.

Regardless of the method chosen, it is my opinion that the monitoring should continue through the first fuel cycle. A decision should be made by the NRC staff at the first refueling outage regarding the need to continue with the monitoring.

Stud-to-Stud Cracks in the Cylinder Block Top

Q21. Do you consider that the qualification test performed on the EDG 103 engine provides an appropriate basis for predicting the behavior of block top cracks in the EDG 101 and 102 engines?

A21. (Bush) No. Differences in the mechanical properties of the cast iron used in the EDG 101 and 102 blocks from the cast iron used in the replacement EDG 103 block and, perhaps more importantly, design changes incorporated into the top of the replacement EDG 103 block do not permit an extrapolation of test results from the latter block to the blocks of EDGs 101 and 102.

Q22. What are your views on the probability that stud-to-stud cracks could initiate in either EDG 101 or EDG 102 during a LOOP/LOCA and propagate to the extent that either engine would be lost from service?

A22. I consider loss of function of EDGs 101 and 102 under these postulated circumstances to be highly improbable for the following reasons:

1. There is no evidence of stud-to-stud cracking in these blocks from previous operation at and above 3500 kW. Such cracks would be more likely to initiate at these higher loads than at the qualified load of 3300 kW.

2. All future surveillance testing is to be accompanied by monitoring of the block tops of EDGs 101 and 102 to verify the continued absence of detectable stud-to-stud cracks.

3. Based on extrapolations from the original EDG 103 block, I would not expect the fatigue crack growth rates in the stud-to-stud area to be so high that there would be a loss of EDG function during a LOOP/LOCA, assuming crack initiation occurred shortly after the start of the LOOP/LOCA. This is particularly true at the low power levels--less than 3000 kW--characteristic of predicted load profiles through most of a LOOP/LOCA, even if one assumes the improbable situation that the engines would be the only source of emergency power for approximately a week. A quantification of crack initiation and growth to the point of loss of function would require a three-dimensional finite element analysis in which crack initiation is assumed. FaAA has conducted such an analysis (FaAA-84-9-11.1, December 1984). My own semi-quantitative assessment is that the cumulative probability of crack initiation and propagation to the point of loss-of-function is quite low.

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'85 FEB -7 P1:31

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
LONG ISLAND LIGHTING COMPANY	)	Docket No. 50-322-OL-4
	)	
(Shoreham Nuclear Power Station, Unit 1)	)	

NRC STAFF TESTIMONY OF JOHN L. KNOX ON SUFFOLK COUNTY AND THE STATE OF NEW YORK EMERGENCY DIESEL GENERATOR LOAD CONTENTION A (i) AND A (iv)

Q. What is your name?

A. My name is John L. Knox.

Q. What is your position?

A. I am a Senior Electrical Engineer (Reactor Systems) in the Power Systems Branch in the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. In this position I perform technical reviews, analyses, and evaluations of reactor plant features pursuant to the construction and safe operation of reactors.

Q. What are your qualifications?

A. In 1962, I received an Associate of Arts degree in Electrical Power System Technology from Montgomery College. In 1971, I received a Bachelor of Science degree in Electronic Systems Engineering from the University of

Maryland. Since 1974, I have taken a number of courses on PWR and BWR system operation, equipment qualification, and reactor safety.

From 1971-1974, I worked for Potomac Electric Company in Washington, D. C. I was assigned to the underground power Transmission Engineering Group and my duties included relocation and restoration of underground power and transmission cables due to the subway construction project. (Prior to this, I spent four years in the Air Force working on the F4 aircraft electronic weapons control systems.)

From 1974 to the present, I have worked for the Nuclear Regulatory Commission involved in the technical review of electrical systems (onsite and offsite power, instrumentation and control). Through 1976, I was a member of the Electrical Instrumentation and Control Systems Branch. This branch was split in January 1977 into an I&C branch and a power branch. Since this split, I have been a member of the Power Systems Branch. My present responsibilities include review and evaluation of onsite and offsite electric power systems.

Q. What is the purpose of your testimony?

A. The purpose of this testimony is to respond to Suffolk County and the State of New York emergency diesel generator load contention a (i) and a (iv), which are as follows:

Contrary to the requirements of 10 CFR Part 50, Appendix A, General Design Criterion 17 -- Electric Power Systems, the emergency diesel generators at Shoreham ("EDGs") with a maximum "qualified load" of 3300 kW do not provide sufficient capacity and capability to assure that the requirements of clauses (1) and (2) of the first paragraph of GDC 17 will be met, in that

(a) LILCO's proposed "qualified load" of 3300 kW is the maximum load at which the EDG may be operated, but is inadequate to handle the maximum load that may be imposed on the EDGs because:

(i) intermittent and cyclic loads are excluded;

(iv) operators may erroneously start additional equipment;

Q. Define the safety function of the emergency diesel generators at Shoreham.

A. The emergency diesel generators are part of the onsite electric power system and as such their safety function was derived from the first paragraph of criterion 17 of Appendix A to 10 CFR 50. The onsite emergency diesel generators "shall be provided to permit functioning of structures, systems, and components important to safety. ...[and] shall...provide sufficient capacity and capability to assure..." this function.

Q. How does the staff determine that the emergency diesel generators have sufficient capacity and capability to perform their safety function?

A. The staff reviews the plant's design loads to ensure that they do not exceed the capacity and capability of the diesel generators.

Q. Define the plant's design load.

A. The plant's design load, as defined in Section 3.4 of IEEE Standard 387-1977, consists of a combination of electric loads, having the most severe power demand characteristic, which is provided with electric energy from a diesel generator unit for the operation of engineered safety features and other systems required during and following shutdown of the reactor.

Q. How can one ensure that the emergency diesel generators have sufficient capacity and capability to perform their safety function?

A. Diesel generator capacity and capability is verified through qualification, preoperational, and periodic testing.

Q. Describe industry recommended practice with respect to load capability qualification testing of diesel generators?

A. Load capability qualification testing as described in IEEE Standard 387-1977 includes, in part, operation of one diesel generator for 22 hours at its continuous rating followed by 2 hours of operation at its short time rating.

Q. Describe the load capability qualification testing performed at Shoreham?

A. Testing at Shoreham included operation of the diesel generator at a 3300 kW load for 750 hours.

Q. Is the 3300 kW load used during the load capability qualification test greater than the plant's design load?

A. Yes, except for intermittent and cyclic loads as indicated on Table 8.3.1-1 and 8.3.1-1A of the FSAR.

Q. What has been estimated to be the worst case kW magnitude and time duration loading for these intermittent and cyclic loads?

A. By letter dated November 19, 1984, the applicant identified the following loads that are automatically actuated, are intermittent/noncontinous, and are not considered to be part of the 3300 kW load used during qualification testing.

a. diesel generator air compressor (12 kW)

b. diesel generator fuel oil transfer pump (0.4 kW)

c. motor operated valves (65.7 kW)

Based on information presented in Table 8.3.1-1 of revision 34 to the FSAR, the staff concludes that the worst case maximum coincident demand of these loads will be 78.1 kW, which, when added to the total maximum emergency service loads tabulated in Table 8.3.1-1A of revision 34 to the FSAR, results in a maximum load of 3331.4 kW. Because the majority of those loads are automatically actuated motor operated valves, they are short duration loads on the order of one to three minutes. Also, automatic actuated valves do not operate simultaneously; therefore, the actual diesel generator loading should be less than the aggregate value of 3331.4 kW but may be greater than 3300 kW for one to three minutes.

In order for each diesel generator to reach its required design basis voltage and frequency limits within the required time of ten seconds, the diesel engine's fuel rack position or fuel setting will move to the wide open position. This wide open fuel setting is greater than the fuel setting which would exist when the diesel generator is delivering steady state power at 3300 kW load. Thus, during this ten second plus time period, the diesel engine may be loaded such that its BMEP may be greater than that corresponding to a continuous electrical load of 3300 kW. Similarly, when individual loads or a block of loads are connected to the generator, the diesel engine's fuel setting will move towards the wide open position. This fuel setting movement maintains the frequency of the generator within the required limits specified in R.G. 1.9. Even though the output of the generator is less than 3300 kW, the diesel engine will be loaded for a short time such that its BMEP may be greater than that corresponding to a continuous electrical load of 3300 kW.

Based on the above, the worst case loading has been estimated to be 3900 kW for less than 60 seconds. The ability of the engines to handle all of the above loads is treated elsewhere in the staff testimony.

Q. It was stated above that diesel generator capacity and capability is verified through qualification, preoperational, and periodic testing. Is the 3300 kW load capability of the diesel generators verified as part of preoperational and periodic testing?

A. Yes.

Q. Describe these tests.

A. As part of the preoperational and 18 month periodic surveillance testing each diesel generator will be operated at 3300 kW for 24 hours. In addition, as part of 30 day periodic surveillance testing, each diesel generator will be loaded to 3300 kW for one hour.

Q. Will the diesel generator's capability to supply intermittent and cyclic loads be verified as part of preoperational and periodic testing?

A. Yes.

Q. Describe these tests.

A. As part of the preoperational and 18 month periodic surveillance testing, each diesel generator will be subject to a load acceptance test. The load acceptance test should demonstrate the capability of each diesel generator to accept the individual loads that make up the plant's design load in the required sequence and time duration. Because intermittent and cyclic loads are part of the plant's design load, the diesel generator's capability to supply these loads should be verified by this test. In addition, as part of six month periodic surveillance testing, each diesel generator will be started within 10 seconds and loaded to 3300 kW within 60 seconds. For this test, the design loads are unavailable for connection to the diesel generator due to the operating mode of the plant. However, this test has been designed to simulate, as close as is practical, the plant's design load. Because the majority of intermittent and cyclic loads will be simulated, the diesel generator's capability to supply these loads will, in part, be verified.

Q. How can this 3300 kW loading, for which the diesel generator has been qualified and is to be periodically tested, be exceeded?

A. The total load that is connectable to the diesel generator exceeds this 3300 kW test loading. Table 8.3.1-1 of the Shoreham FSAR indicates that the total connectable loads are 4381.3 kW for diesel generator number 101, 4147.8 kW for diesel generator number 102, and 4493.7 for diesel generator number 103. These loads could be connected manually or by equipment failure.

In LILCo testimony of G. F. Dawe, J. A. Notaro, and E. J. Youngling on pages 32 through 35, it was stated that the single worst case load that could be started erroneously as a result of an operator error following a LOOP/LOCA would result in the following loads on the diesel generators:

1. 3459.4 kW on DG 101
2. 3414.8 kW on DG 102
3. 3583.5 kW on DG 103

The single worst case load that could be started erroneously as a result of an operator error following a LOOP would result in the following loads on the diesel generator:

1. 3839.2 kW on DG 101
2. 3627.6 kW on DG 102
3. 3867.3 kW on DG 103

Q. How does the staff normally ensure that diesel generators have sufficient capacity and capability to handle intermittent/cyclic loads and additional loads that may be inadvertently connected to the diesel generator by operator error or equipment failure?

A. The staff ensures that the diesel generator has a two-hour short-term overload capability which encompasses these loads.

Q. Do the Shoreham diesel generators have an overload rating?

A. No. The 3300 kW qualified load rating is the only rating. The ability of the diesel generator to handle loads above 3300 kW is addressed elsewhere.

Q. Should diesel generators used for nuclear service have an overload rating in order to meet the capacity and capability requirement of Criterion 17?

A. Yes

Q. Why?

A. To ensure that the diesel generators have sufficient capacity and capability to supply the plant's design loads which include intermittent/cyclic loads and additional loads that may be inadvertently connected to the diesel generator by operator error or equipment failure.

Q. What provisions has LILCO proposed to prevent the 3300 kW loading from being exceeded?

A. LILCO has proposed procedures and training changes with a plant technical specification limit of 3300 kW on each diesel generator. The adequacy of procedures is addressed elsewhere in the staff's testimony.

Q. Describe what a 3300 kW technical specification limit on the diesel generator means?

A. As part of the Shoreham technical specifications, a 3300 kW maximum limit on each diesel generator will be imposed as a condition to the Shoreham license. If 3300 kW is exceeded at any time by any amount the associated technical specification action will require the plant to be shut down with a subsequent analysis and inspection performed to demonstrate the capability of the diesel generator before continued plant operation would be allowed. In addition, the calibration of the instrumentation used to monitor kW output of each diesel generator will be included in the Shoreham technical specifications.

Q. With these provisions proposed by LILCO, does one have reasonable assurance that disabling overloading of the diesel generators will be prevented during transient and accident conditions?

A. Yes, provided the diesel generator is qualified for the expected overloading during transient and accident conditions and for expected operation at 3300 kW following overloading. The qualification of the diesel generator is addressed elsewhere in the staff's testimony.

Q. In addition to these administrative provisions proposed by LILCO, what else would LILCO have to do to provide reasonable assurance that the diesel generators have sufficient capacity and capability to perform their safety function and meet the requirements of criterion 17 of Appendix A to 10 CFR 50.

A. LILCO must demonstrate that their diesel generators are qualified for an acceptable short-term overload capability as part of preoperational and 18-month periodic surveillance testing.

Q. What would be the magnitude and duration of loads for which the diesel generator would need to be qualified and periodically tested?

A. Design load analyzed for the Shoreham plant plus the sum of the following overloads:

1. A load equal to the worst case loading that could be connected to any one diesel generator by a single operator error or event, plus
2. A load or sum of loads that are to be added or connected to the diesel generator intentionally according to the plant procedures.

February 5, 1985

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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

CARL H. BERLINGER

on

LOAD CONTENTIONS CONCERNING TDI EMERGENCY DIESEL GENERATORS

at the

SHOREHAM NUCLEAR POWER STATION

Q. Please state your name and occupation.

A. My name is Carl H. Berlinger. I am employed by the U. S. Nuclear Regulatory Commission and I am currently assigned as the Manager of the TDI Project Group in the Division of Licensing.

Q. What are your qualifications and experience relevant to your testimony?

A. A copy of my professional qualifications and experience have been previously submitted as an attachment to an affidavit filed by the NRC on February 16, 1984.

Q. What is the purpose of your testimony?

A. This testimony is for purposes of stating that the joint testimony filed by our consultant/contractor, Battelle, Pacific Northwest Laboratory has been reviewed by the NRC staff and that their testimony has been accepted for filing on behalf of the NRC staff.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

CERTIFICATE OF SERVICE

I hereby certify that copies of the testimony of John Knox, Wayne Hodges, Carl Berlinger, James Clifford, Joseph J. Buzy, Richard J. Eckenrode, S. H. Bush, A. J. Henrikson, and A. Sarsten in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system, this 5th day of February, 1985.

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