UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 6, 1988

Docket No. 50-322

Mr. John D. Leonard, Jr.
Vice President-Nuclear Operations
Long Island Lighting Company
Shoreham Nuclear Power Station
P.O. Box 618, North Country Road
Wading River, New York 11792

Dear Mr. Leonard:

SUBJECT: STAFF'S TECHNICAL EVALUATION OF LILCO'S REQUEST TO OPERATE THE SHOREHAM NUCLEAR POWER STATION AT TWENTY-FIVE PERCENT POWER (TAC NO. M65085)

RE: SHOREHAM NUCLEAR POWER STATION

On April 14, 1987, the Long Island Lighting Company (LILCO) submitted to the Commission a request for authorization to increase power up to 25 percent of full-rated power at the Shoreham Nuclear Power Station (SNPS). This request was made pursuant to the provisions of 10 CFR 50.47. Specifically, LILCO contends that the implementation of its emergency plan by its local emergency response organization and local governments on a best-efforts basis, coupled with the 25-percent power limitation, constitutes an adequate compensating measure for the interim period while the contested emergency planning (EP) issues are being litigated for full-power operation.

Enclosed is a technical evaluation of LILCO's request. This evaluation addresses the following three categories of issues related to operation at a reduced power level:

1. Systems and Procedures for Accident Mitigation
2. Accident Evaluation
3. Safety of Prolonged Operation at Twenty-Five Percent Power

On the basis of its evaluation, the staff has determined that (1) the current SNPS systems and procedures are adequate for accident mitigation for operation at 25 percent power, (2) the times available to take corrective actions for accidents are increased and the consequences of accidents are reduced at the lowered operating power levels, and (3) SNPS can operate safely for prolonged periods of time at the lower operating power levels with an augmented inspection program for certain components.

DF01
1/1

If you should have any questions about the enclosed safety evaluation, please contact Mr. Stewart Brown, the staff's Shoreham Project Manager, at (301) 492-1444.

Sincerely,

/S/

Steven A. Varga, Director
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

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RPalla	TMarsh	RBarrett
WHodges		

Previously concurred*

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RLo	AThadani	JRichardson	FCongel	MYoung	SVarga
07/18/88	09/19/88	07/29/88	09/15/88	09/22/88	/88

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PDII-1*	NRR/SAD	NRR/DEST;AD/E*	NRR/DREP	OGC/HFB	DRP/DRI
RLo	AThadani	JRichardson	FCongel	[illegible]	SVarga
07/18/88	/ /88	07/29/88	/ /88	9/22/88	/ /88

comments included
10/5/88

If you should have any questions about the enclosed safety evaluation, please contact Mr. Stewart Brown, the staff's Shoreham Project Manager, at (301) 492-1444.

Sincerely,

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comments incorporated SS 9/19

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Sincerely,

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07/18/88	/ /88	07/29/88	9/1/88	/ /88	/ /88

If you should have any questions about the enclosed safety evaluation, please contact Mr. Stewart Brown, the staff's Beaver Valley Project Manager, at (301) 492-1444.

Sincerely,

Steven A. Varga, Director
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

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If you should have any questions about the enclosed safety evaluation, please contact Mr. Stewart Brown, the staff's Shoreham Project Manager, at (301) 492-1444.

Sincerely,

Steven A. Varga, Director
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

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If you should have any questions about the enclosed safety evaluation, please contact Mr. Stewart Brown, the staff's Shoreham Project Manager at (301) 492-1444.

Sincerely,

Steven A. Varga, Director
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If you should have any questions about the enclosed safety evaluation, please contact Mr. Stewart Brown, the staff's Shoreham Project Manager at (301) 492-1444.

Sincerely,

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* Cover Letter Enclosures

If you should have any questions about the enclosed safety evaluation, please contact Mr. Stewart Brown, the staff's Shoreham Project Manager, at (301) 492-1444.

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Division of Reactor Projects I/II
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If you should have any questions about the enclosed safety evaluation, please contact Mr. Stewart Brown, the staff's Shoreham Project Manager at (301) 492-1444.

Sincerely,

Steven A. Varga, Director
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

THE NRC STAFF'S TECHNICAL REVIEW OF A REQUEST FROM LONG ISLAND LIGHTING COMPANY FOR AUTHORIZATION TO OPERATE THE SHOREHAM NUCLEAR POWER STATION AT A POWER LEVEL UP TO TWENTY-FIVE PERCENT OF FULL-RATED POWER

INTRODUCTION

On April 14, 1987, the Long Island Lighting Company (LILCO) submitted to the Commission a request for authorization to increase power to 25 percent of rated power at the Shoreham Nuclear Power Station (the Request). In the Request, LILCO claims that 10 CFR 50.47 provides the regulatory basis for the authorization for operation beyond 5 percent power, despite the existence of unresolved emergency planning (EP) contentions. In the Request, LILCO seeks to demonstrate that it can meet all three of the conditions set forth in 10 CFR 50.47(c)(1) with the restriction on power level to 25 percent. Specifically, LILCO contends that the implementation of its emergency plan by its local emergency response organization and local governments on a best-efforts basis, coupled with the 25 percent power limitation, constitutes an adequate compensating measure for the interim period when the contested EP issues are still being litigated in regard to the full-power license.

In terms of the power limitation as an "interim compensating action," LILCO claims that the risk and consequences of accidents at 25 percent power operation are so greatly reduced that the remaining unresolved EP issues become insignificant. The staff's technical review is an attempt to assess the validity of LILCO's claim about this reduction of risks and consequences from the analysis that was submitted with the Request. The staff's evaluation does not examine the unresolved EP issues and whether the safety merits associated with the 25 percent power restriction would constitute adequate compensating measures. Instead, the emphasis of this technical review is on comparisons between operation at 25 percent power and at full power and the effects of the power reduction on various aspects of postulated accidents.

On September 9, 1988, the Federal Emergency Management Agency (FEMA) provided its finding on LILCO's offsite emergency response plan for Shoreham. FEMA stated that the full participation exercise conducted on June 7-9, 1988 demonstrated adequate overall preparedness on the part of Local Emergency Response Organization personnel. Therefore, based on the evaluation of the plan and the exercise, FEMA reached a finding of reasonable assurance that the health and safety of the public living in the vicinity of the plant can be protected. FEMA's plan review and exercise evaluation were based on the assumptions that in an actual radiological emergency, State and local officials that have declined to participate in emergency planning will (1) exercise their best efforts to protect the health and safety of the public, (2) cooperate with the utility and follow the utility plan, and (3) have the resources sufficient to implement those portions of the utility plan where State and local response is necessary.

SCOPE OF STAFF'S REVIEW

The staff's review addresses the following three categories of issues:

(1) Systems and Procedures for Accident Mitigation
(2) Accident Evaluation
(3) Safety of Prolonged Operation at Twenty-Five Percent Power

(1) Systems and Procedures for Accident Mitigation

Except for the questions related to EP, all safety issues have been satisfactorily resolved for full-power operation. In its analysis to demonstrate that there are reduced risk and accident consequences when operating at 25 percent power compared with operating at full-power, LILCO cites several physical and procedural improvements made at the Shoreham Nuclear Power Station (SNPS) since the issuance of its 5 percent power license in July 1985. The staff has reviewed the acceptability of these hardware and procedural changes for the credit taken in the accident analysis in support of the Request. The Safety Evaluation prepared by the staff is provided as Enclosure 1.

(2) Accident Evaluation

The design basis accidents (DBAs) for full-power operation were addressed in Section 15 of the SNPS Final Safety Analysis Report, and the consequences of these accidents would not result in the need for offsite evacuation. Therefore, only those accidents that are beyond the DBA need to be evaluated. To support its request, LILCO presented a probabilistic risk analysis (PRA) to show that at 25 percent power (a) the probabilities of core-melt accidents are reduced; (b) the offsite radiological consequences of accidents are reduced; and (c) the timing for key events in the accident progression, e.g., core slump, reactor vessel failure, and releases to the environment, is significantly increased. This consideration is beneficial in two important aspects: first, the available time enhances the opportunity for corrective actions (e.g., correct diagnostics, restoration of core cooling, restoration of ac power) to arrest the accident progression, and secondly, the increased duration between the onset of an accident and releases of radioactive materials to the environment will significantly increase the time available for emergency responses. The staff's review of LILCO's PRA-based portion of the request is provided as Enclosure 2.

(3) Safety of Prolonged Operation at Twenty-Five Percent Power

Prolonged off-normal operation at 25 percent power may cause instability or other undesirable effects on certain safety-related systems. The staff performed an evaluation concerning the reliability of those systems and equipment for which performance is identified to be power-level dependent. The Safety Evaluation on this issue is provided as Enclosure 3.

SUMMARY OF RESULTS

The following is a summary of the significant results of the staff's evaluation as presented in Enclosures 1, 2, and 3.

(1) Systems and Procedures for Accident Mitigation

The staff finds the following improvements in equipment and procedures to be acceptable for the credits taken in the risk assessment:

(1) The main condenser as the viable heat sink following a turbine trip. The majority of anticipated transient initiators for boiling water reactors (BWRs) result from or lead to a turbine trip. With the 25 percent power limitation, availability of the main condenser as the only necessary heat sink is an important mitigating factor for anticipated transient without scram (ATWS) accidents.

(2) The standby liquid control system at SNPS, which is designed to ensure an equivalent boron injection capability that is 200 percent of the ATWS rule requirement of 10 CFR 50.62.

(3) Compliance with the ATWS rule for the alternate rod injection and recirculation pump trip capabilities to mitigate ATWS accidents.

(4) The design and installation of the "corium ring," which is intended to channel the molten core debris (corium) directly into the suppression pool for quenching, even though this hardware modification is not explicitly modelled in the risk assessment.

(5) The additional AC power supplies that are beyond those installed to meet the requirements of Criterion 17 of Appendix A to 10 CFR Part 50. These additional ac power supplies include the gas turbine, four mobile diesel engines, and the Colt diesel-engine-powered generators that could mitigate or avert station blackout accidents.

(6) The procedure to use the diesel fire pump as a viable cooling source.

(7) The availability of operator options to gain greater control in accident mitigation actions (e.g., throttling of the low-pressure emergency core cooling system and the condensate flow during ATWS events, the capability to switch the high-pressure coolant injection suction to either the suppression pool or the condensate storage tank, and the enhancement of the automatic depressurization system initiation logic).

(2) Accident Evaluation

The staff's review of LILCO's PRA-based accident analysis for 25 percent power operation concentrates on comparisons with 100 percent power operation. These comparisons were to determine the validity of LILCO's claim that 25 percent power operation involves significant improvements in terms of vulnerability to

core damage accidents, additional time to respond to accidents, and reductions of offsite consequences of the postulated accidents. These comparisons were based on the use of the same calculation tools and assumptions and have an advantage over calculations for absolute values because the effects of inherent modelling uncertainties tend to be minimized. Furthermore, the comparisons of the timing of events during an accident and of offsite consequences are deterministic in nature, given an assumed core damage accident progression sequence or assumed release characteristics. The uncertainties associated with the probabilities of the paths of the accident development are not relevant for these deterministic comparisons.

(a) Vulnerability to Core Damage Accidents

The LILCO 25 percent power PRA calculated a core-melt frequency reduction of approximately a factor of two due to 25 percent operation, in conjunction with improvements in plant design and emergency operating procedures. The calculated reduction was about a factor of three for station blackout and ATWS sequences.

For a number of specific plant vulnerabilities to core melt, the staff's evaluation found that the 25 percent power restriction, in conjunction with improvements in plant design and emergency operating procedures, represents improvements that are in general agreement with LILCO's claim. The staff's evaluation supports LILCO's claim that the overall core melt frequency is reduced at 25 percent power compared with 100 percent power. However, the staff did not verify the absolute magnitude of this reduction.

(b) Offsite Consequences of Accidents

The staff's evaluation found LILCO's claim that offsite radiological consequences are reduced at 25 percent power vs. 100 percent power operation to be valid. This is due to two factors: first, a reduction of an approximate factor of four in the fission product inventory available for release in any postulated accident at 25 percent power; second, a significant increase in the time to release because of the reduced heatup rate at 25 percent power, i.e., the reduced decay heat level.

The staff's evaluation is in agreement with LILCO's analysis that indicates that there is considerable reduction in the probability of exceeding a given dose at an offsite location, even without evacuation. This is particularly the case for larger doses. The distances over which injury-threatening doses (i.e., 200 rem) would occur are reduced by a factor of about three compared with 100 percent power operation. The staff has performed dose-distance calculations for both 25 percent and 100 percent power using the same code and assumptions. For the sequences representing the bulk of the core-melt accidents at Shoreham the calculated probability of exceeding 200 rem falls off rapidly to small values at distances of about one mile from the SNPS site at 25 percent power, versus about three miles at 100 percent power.

For the less probable, rapidly evolving accidents, representing about three percent of the core-melt frequency in the Request, the calculated probability of exceeding 200 rem falls off rapidly to small values about two miles from the site at 25 percent power compared with 10 miles at 100 percent power.

Another important consideration of the accident consequence calculation is the significant additional time available to avoid injury-threatening doses. Dose-versus-time calculations performed for a rapidly evolving accident sequence show that a 200-rem whole-body dose would not be reached at a two mile radius within six hours after accident initiation, in comparison to about one hour in the case of 100 percent power operation.

(c) Timing of Accident Progression

The staff's evaluation agrees with LILCO's claim that operation at 25 percent power would result in considerable delay in accident progression when compared with similar accidents occurring at 100 percent power operation. The staff found that significant delays would occur in all postulated accident sequences and at every stage of accident development. The major mitigating factor is the 25 percent power limitation and the associated reduction of decay heat that is the driving force in accident progression.

For a large group of accidents, characterized by delayed challenges to the containment integrity, releases to the environment at 25 percent power would not occur until well over 12 hours after accident initiation. These accidents include those initiated by a loss of offsite power, the majority of loss-of-coolant accidents, and those transient-initiated sequences for which the reactor is successfully shutdown but core cooling is inadequate. These accidents contribute over 80 percent of the total core-melt frequency. Under more optimistic assumptions regarding reactor vessel failure, core-concrete interactions, and containment performance, the time of releases to the environment for these sequences at 25 percent power would be on the order of a day or more. For 100 percent power operation, these accidents generally lead to radioactive releases in the order of several hours (the majority in the four-to-seven-hour range).

The most rapidly developing accidents, where contribution to overall estimated core melt frequency is very small, are those characterized by early containment failure or containment bypass releases. The dominant accident in this category is the seismically induced accident that breaches the reactor coolant boundary as well as the containment. The staff estimates that the time from the onset of the accident to the time when radioactive releases to the environment occur is about one hour. The corresponding time estimated for 100 percent power is about ten minutes.

The remaining category of accidents is dominated by those involving transients with failure to scram the reactor. The staff estimates that radioactive releases to the environment for this category of accidents for the 25 percent power case is about seven to 12 hours from the onset of the transient. For the case of 100 percent power operation, the higher decay heat represents an earlier challenge to the containment integrity, and releases are estimated to occur in about two and one half hours.

(3) Safety-Related Systems Evaluation

The staff agrees with LILCO's evaluation that all safety-related equipment is intended to be operated over the entire power range. However, the staff is concerned if reduced power operation would cause accelerated wear or early fatigue damage to certain safety-related equipment from low-flow-induced vibration or instability. The staff found three systems--the reactor recirculation, the main steam, and the feedwater systems--to be power dependent and to operate at a reduced flow. In particular, the feedwater check valves are most vulnerable and would serve as a good indicator of any potential equipment deterioration. The staff determines that these check valves should be subject to a more frequent inservice testing inspection schedule; that is, these valves should be inspected during each refueling outage but at least every two years, if refueling outages are longer.

CONCLUSION

The following are the major findings of the staff's evaluation for 25 percent power operation at SNPS:

(1) There are no new unresolved safety questions associated with 25 percent power operation that have not been analyzed during the full-power licensing review process.

(2) The improvements in equipment and procedures are acceptable for the credits taken in the accident analysis as presented in the Request.

(3) The staff is in general agreement with LILCO's claim that operation at 25 percent power would reduce core-melt frequency.

(4) There are significant delays in the time of progression for all the postulated accident sequences when compared with those during 100 percent power operation:

   (a) For accidents contributing to about 80 percent of the 25 percent power core-melt frequency, a long time is required for core melt and vessel failure, and radioactive releases would not occur in less than 12 hours.

   (b) The most rapidly developing accidents are those associated with a rapid loss of coolant and failure of all injection systems. A seismic event that breaches the reactor coolant system and the containment is a representative sequence of this type. The probability of these accidents occurring is small, accounting for about three percent of core-melt frequency in the LILCO PRA for 25 percent power. The onset of releases are delayed from about 12 minutes in the case of 100 percent power operation to an hour at 25 percent power. The bulk of radiological releases would occur later: one hour at full power and three hours at 25 percent power.

(c) The remaining category of accidents in terms of adequate timing to take mitigating action is dominated by ATWS sequences. Radioactive releases to the environment for these accidents are estimated to occur in about seven to 12 hours. This compares with an estimate of about two and one half hours for similar accidents during 100 percent power operation.

(5) The equilibrium radionuclide inventories are reduced by about a factor of four when compared with 100 percent power operation; offsite radiological consequences can also be expected to be reduced by the same factor.

(6) The distances from the SNPS site within which injury-threatening radioactive doses could occur without evacuation have been significantly reduced. In staff calculations performed with the same code and assumptions, the staff found that these distances have been reduced to one or two miles for 25 percent power operation. While the vulnerable areas have been reduced, time available for evacuation from the reduced areas has been significantly increased in the case of 25 percent power operation at SNPS over 100 percent power.

(7) Certain plant components could be adversely effected by prolonged operation at a reduced power level due to reduced system flow condition. These components should be subject to a more frequent inservice testing program than required for 100 percent power operation.

REFERENCE

Letter to Victor Stello (NRC) from Grant Peterson (FEMA), dated September 9, 1988.

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