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

LONG ISLAND LIGHTING COMPANY

DOCKET NO. 50-322

SHOREHAM NUCLEAR POWER STATION

FACILITY OPERATING LICENSE

License No. NPF-36

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for a license filed by the Long Island Lighting Company (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Shoreham Nuclear Power Station (the facility), has been substantially completed in conformity with Construction Permit No. CPPR-95 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission (except as exempted from compliance in Section 2.D. below);

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below);

   E. The licensee is technically qualified to engage in the activities authorized by this operating license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. The licensee has satisfied the applicable provisions of 10 CFR 140, "Financial Protection Requirements and Indemnity Agreements", of the Commission's regulations;

G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Facility Operating License No. NPF-36, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Based on the foregoing findings; the Partial Initial Decision issued by an Atomic Safety and Licensing Board on September 21, 1983; the Atomic Safety and Licensing Appeal Board Decision (ALAB-788) dated October 31, 1984; Commission Memorandum and Order CLI-84-21, dated November 21, 1984; the Memorandum and Order Ruling on Remand Issues dated November 30, 1984, issued by an Atomic Safety and Licensing Board regarding this facility; and the Partial Initial Decision on Emergency Diesel Generators issued by an Atomic Safety and Licensing Board on June 14, 1985; Facility Operating License No. NPF-36 is hereby issued to the Long Island Lighting Company (the licensee) to read as follows:

A. This license applies to the Shoreham Nuclear Power Station, a boiling water nuclear reactor and associated equipment, owned by the licensee. The facility is located in Suffolk County, New York, and is described in the licensee's Final Safety Analysis Report, as supplemented and amended, and the licensee's Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Long Island Lighting Company (LILCO, the licensee):

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use, and operate the facility at the designated location in Suffolk County, New York, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect except as exempted from compliance as described in Section 2.D. below; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at core power levels not to exceed 121.8 megawatts thermal (5% power).

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Fire Protection Program (Section 9.5, SER, SSER1, SSER2, SSER3)

a. The licensee shall maintain in effect all provisions of the approved fire protection program as described in the Fire Hazards Analysis Report and the Final Safety Analysis Report for the facility through Revision 33 and as approved in the SER through Supplement 3, subject to provisions b and c below.

b. The licensee may make no change to features of the approved fire protection program which would decrease the level of fire protection in the plant without prior approval of the Commission. To make such a change the licensee must submit an application for license amendment pursuant to 10 CFR 50.90.

c. The licensee may make changes to features of the approved fire protection program which do not decrease the level of fire protection without prior Commission approval, provided:

(i) such changes do not otherwise involve a change in a license condition or technical specification or result in an unreviewed safety question (see 10 CFR 50.59).

(ii) such changes do not result in failure to complete the fire protection program approved by the Commission prior to license issuance.

The licensee shall maintain, in an auditable form, a current record of all such changes including an analysis of the effects of the change on the fire protection program and shall make such records available to NRC inspectors upon request. All changes to the approved program made without prior Commission approval shall be reported annually to the Director of the Office of Nuclear Reactor Regulation together with supporting analyses.

(4) Initial Test Program (Section 14, SER, SSER1)

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Inservice Inspection and Testing Program (Section 5.2.4 SER, Section 6.6 SER, SSER1, SSER4, Section 4.5.2, SSER7)

a. The initial inservice inspection program will be evaluated before the first refueling outage (reference SER Sections 5.2.7, 6.6, SSER1 Sections 5.2.7, 6.6, SSER4 Section 6.6).

b. The development of the Shoreham ISI program shall incorporate provisions involving (1) the use of Monticello-type techniques for the detection of intergranular stress corrosion cracking, and (2) an inspection program scope consistent with that in Section 5.2.3.2.1.3 of the Preliminary Safety Analysis Report for the Perry plant (Docket 50-440). The licensee shall also notify the NRC staff of any significant or substantive changes in the intended inspection program, and shall continue to evaluate and implement, where practicable, state-of-the-art improvements in scope or methods of implementing the ISI program. (Section 4.5.2, SSER7).

(6) Surveillance of Control Blades (Section 4.1.3.14, SER)

Within 30 days after plant startup following the first refueling outage, the licensee shall comply with items 1, 2, and 3 of IE Bulletin No. 79-26, Revision 1, "Boron Loss from BWR Control Blades", and submit a written response on Item 3.

(7) NUREG-0737 Conditions (Section 22, SER)

The licensee shall complete the following conditions to the satisfaction of the NRC. These conditions reference the appropriate items in Section 22.2, "TMI Action Plan Requirements for Applicants for Operating Licenses," in the Safety Evaluation Report and Supplements 1 and 4, of NUREG-0420.

(a) Shift Technical Advisor (Section 22.2 (Item I.A.1.1) SSER1)

The licensee shall submit the qualifications each backup shift technical advisor (STA) is expected to have at the completion of their training program, for review and approval by the NRC staff prior to assigning them to STA duty. This license condition shall terminate upon the completion of NRC staff approval of the first group of seven backup STAs.

(b) Control Room Design Review (Section 22.2 (Item I.D.1) SSER1, SSER3)

Prior to completion of the startup test program, appropriate control room meters and recorders shall be marked to indicate normal operating limits, trip values and alarm points.

(c) Post Accident Sampling Capability (Section 22 (Item II.B.3) SSER4)

Prior to startup following the first refueling outage, the licensee shall submit to the NRC staff a modified core damage procedure that includes an estimation of cladding failure due to fuel overheating, as well as cladding failure and core melt for review and approval. This procedure shall incorporate the use of other plant parameters as indicators of core damage.

(d) Emergency Response Capabilities (Generic Letter 82-33, Supplement 1 to NUREG-0737)

The licensee shall complete emergency response facilities and capabilities as required in Attachment 1 of this license.

(8) Equipment Qualification (Section 3.11, SSER7)

Prior to November 30, 1985 the licensee shall environmentally qualify all electrical equipment according to the provisions of 10 CFR 50.49.

(9) Instrumentation and Controls Systems Required for Safe Shutdown (Section 7.4.3 SSER3, SSER4, SSER 8)

Prior to startup following the first refueling outage, the licensee shall implement and document all of the required design changes discussed in Attachment 4 and shall perform an acceptable procedure verification test for the remote shutdown system design.

(10) Concrete and Structural Steel Internal Structures (Section 3.8.2 SER, SSER1, SSER3, SSER4)

The licensee shall not operate the residual heat removal (RHR) system in the steam condensing mode (SCM) during any normal plant operations. However, the SCM may be used, in accordance with the provisions of 10 CFR 50.54 (x) and (y), as a last resort when all other means of core or containment cooling have been lost.

(11) Containment Isolation System (Section 6.2.3 SER, SSER1, SSER3, SSER4, SSER 8)

Prior to start-up following the first refueling outage the licensee shall install two isolation barriers in series in all instrument lines penetrating containment that are not part of the automatic reactor protection system. Proposals from the licensee on how this shall be accomplished as well as details on the necessary design changes, shall be submitted for NRC staff review and approval.

(12) Emergency Diesel Generator License Condition

The license conditions described in Attachment 3 apply to the operation, maintenance, testing, and inspection of Emergency Diesel Generators EDG-101, 102, and 103 at Shoreham. With the imposition of these conditions, the Commission has determined that the facility satisfies the requirements of General Design Criterion 17 (GDC-17) for the first cycle of operation. Prior to startup following the first refueling outage, the licensee shall submit the results of additional tests, inspections, and/or analyses that taken together with the nuclear industry experience with TDI diesels up to that time, demonstrate to the satisfaction of the NRC staff that the facility will continue to satisfy the requirements of GDC-17 for the second fuel cycle and beyond.

(13) Independent Design Review (IDR) (Section 17.7, SSER 7)

Prior to exceeding five percent of rated power, the licensee shall incorporate the studies and evaluations performed by the licensee or its contractors as a result of the IDR, into the existing plant calculation and documentation packages, in order to provide a complete set of records to be used for maintenance, replacement, repair, and modification of equipment.

(14) Seismic and Dynamic Qualification (Section 3.10, SSER 3, SSER 7, SSER 8)

(a) Prior to exceeding five percent of rated power, the licensee shall complete the qualification, documentation, and installation of:

(1) Radiation Monitoring System Panels (Mark 1D11*PNL-117A and B)

(2) Radiation Monitoring System Pumps (Mark 1D11*P-126, 134)

(b) Prior to its use as an invessel storage area for irradiated fuel bundles, the licensee shall complete qualification and documentation for the invessel rack (F16-E006/1F16 *FAK-09)

(15) Operating Staff Experience Requirements (Section 13, SSER 8)

The licensee shall have on shift operators that meet the requirements described in Attachment 2.

(16) Fission Gas Release and Ballooning and Rupture (Sections 4.2.3.2 and 4.2.3.3 SER)

The licensee shall reanalyze the ECCS performance for the second cycle and beyond utilizing models that (a) account for the effects of high burn-up fission gas release and prepressurized fuel, (b) accommodate the information in NUREG-0630 including its effects on local oxidation, and (c) have been reviewed and approved by the NRC.

(17) Generic Implications of Salem ATWS Events (Generic Letter 83-28)

The licensee shall implement its response to the requirements of Generic Letter 83-28 on a schedule which is consistent with that stated in its letters of March 9, 1984 (SNRC-1013) and December 4, 1984 (SNRC-1116).

D. The facility requires exemptions from certain requirements of Appendices A and J to 10 CFR Part 50. These include: (a) exemption for operation at up to five percent of rated power from General Design Criterion (GDC) 2 of Appendix A, for the seismic qualification of the Radiation Monitoring System Panels (Mark 1 D11*PNL-117 A and B) and Radiation Monitoring System Pumps (Mark 1 D11*P-126, 134), (Section 3.10, SSER 3, SSER 7, SSER 8); (b) exemption from GDC 56, the installation prior to startup following the first refueling outage of two isolation barriers in series in all instrument lines penetrating containment that are not part of the automatic reactor protection system (Section 6.2.3, SER, SSER 1, SSER 3, SSER 4, SSER 8); (c) exemption from the requirement of paragraphs II.H.4 and III.C.2 of Appendix J, the leak rate testing of the Main Steam Isolation Valves (MSIVs) at the peak calculated containment pressure, P_a, and exemption from the requirements of paragraph III.C.3 of Appendix J that the measured MSIV leak rates be included in the summation for the local leak rate (Section 6.2.5.1 of the SER and SSER 8); and (d) exemption until startup after the first refueling outage from GDC 19, that the remote shutdown system design should provide redundant safety-grade capability to achieve and maintain hot shutdown and subsequently cold shutdown from a location or locations remote from the control room, assuming no fire damage to any required systems and equipment and assuming no accident has occurred (Section 7.4.3 of SSER 3, SSER 4, and SSER 8). These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans, including all amendments and revisions made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p), which are part of the license. These plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Shoreham Nuclear Power Station Security Plan," "Shoreham Nuclear Power Station Safeguards Contingency Plan," and the "Shoreham Nuclear Power Station Guard Training and Qualification Plan."

F. Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, the licensee shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within twenty four (24) hours to the NRC Operations Center via the Emergency Notification System with written followup within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

G. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. The conditions incorporated into the license may be subject to further modification or addition as a result of the adjudicatory proceeding which is pending before an Atomic Safety and Licensing Board concerning offsite emergency planning.

I. This license supersedes Facility Operating License No. NPF-19, dated December 7, 1984.

J. This license is effective as of the date of issuance and shall expire at midnight on April 13, 2013.

FOR THE NUCLEAR REGULATORY COMMISSION

Darwin Eisenhut
for Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Attachment 1
2. Attachment 2, NUREG-0737
   Supplement 1 Schedule
3. Attachment 3
4. Attachment 4
5. Attachment 5
6. Appendix A - Technical
   Specifications (NUREG-1126)
7. Appendix B - Environmental
   Protection Plan

Date of Issuance:

JUL 8 1985

ATTACHMENT 1

NUREG-0737, Supplement 1 Schedule

The licensee shall implement the specific items below, in the manner described in letter SNRC-863, dated April 14, 1983, as modified by letter SNRC-1103, dated November 5, 1984, no later than the following dates.

1. Safety Parameter Display System (SPDS)

Final SPDS fully operational and operators trained	Prior to Startup after the first refueling outage 1/

2. Detailed Control Room Design Review (DCRDR)

Submit a Summary Report to the NRC including a proposed schedule for implementation	Prior to startup following the first refueling outage.

3. Regulatory Guide 1.97, Revision 2

Implement the requirements of RG 1.97 or provide justification for deviations	2/

4. Upgrade Emergency Operating Procedures (EOP)

a. Submit a Procedures Generation Package to the NRC	Prior to the start of the first refueling outage.
b. Implement the upgraded EOP's	Prior to startup after the first refueling outage.

1/ Installation of an interim SPDS has been completed.

2/ The licensee has taken an exception to some of the guidance of Regulatory Guide 1.97, Revision 2 with regard to neutron flux, reactor water level, RCS soluble boron concentration, drywell sump level and drywell drain sump level, primary containment isolation valve position, radiation level in circulating primary coolant, analysis of primary coolant radiation exposure rate, suppression chamber spray flow and drywell spray flow, suppression pool water level, core spray system flow, SLCS storage tank level, reactor building area radiation, environs radiation and radioactivity and primary coolant and sump grab sampling. The licensee's proposal is under review by the staff; the licensee shall implement any additional modifications which arise from that review prior to startup following the first refueling unless prior approval of these exceptions is granted by the NRC staff.

ATTACHMENT 2

Operating Staff Experience Requirements (Section 13, SSER8)

The licensee shall have a licensed senior operator on each shift who has had at least six months of hot operating experience on a same type plant, including at least six weeks at power levels greater than 20% of full power, and who has had start-up and shutdown experience. For those shifts where such an individual is not available on the plant staff, an advisor shall be provided who has had at least four years of power plant experience, including two years of nuclear plant experience, and who has had at least one year of experience on shift as a licensed senior operator at a similar type facility. Use of advisors who were licensed only at the RO level will be evaluated on case-by-case basis. Advisors shall be trained on plant procedures, technical specifications and plant systems, and shall be examined on these topics at a level sufficient to assure familiarity with the plant. For each shift, the remainder of the shift crew shall be trained in the role of the advisors. The training of the advisors and remainder of the shift crew, and the assignment of an advisor to each shift, shall be completed at least one week prior to exceeding five percent of rated power. The licensee shall at that time certify to the NRC staff the names of the advisors who have been examined and have been determined to be competent to provide advice to the operating shifts. These advisors shall be retained until the experience levels identified in the first sentence above have been achieved. The NRC staff shall be notified at least 30 days prior to the date the licensee proposes to release the advisors from further services.

ATTACHMENT 3

Emergency Diesel Generator (EDG) 101, 102, and 103 License Conditions

A. Cylinder Blocks

1. During any period of continuous operation following automatic diesel generator initiation, the licensee shall perform daily visual inspections of the area between adjacent cylinder heads and the general block top. The licensee shall also perform visual inspections of the same areas under intense light during the monthly surveillance testing. This condition is applicable to EDGs 101, 102, and 103.

2. Following any loss of offsite power event which results in the automatic starting of any EDG during the first fuel cycle, and during the first refueling outage, the licensee shall inspect the top surface of the block exposed by the removal of two appropriate cylinder heads from each of the three EDG engines. Inspections shall be by liquid penetrant, with eddy current for any identified cracks, to determine the presence of new cracks and the depth of any new or old cracks.

3. The licensee shall perform eddy current testing between adjacent cylinder heads after any operation of EDG 101 or 102 at greater than 1800 KW. If indications of a stud-to-stud crack are detected, the licensee shall declare the associated engine inoperable until an inspection of the area in question has been done and until additional analyses of the condition have been performed and approved by the NRC staff.

4. The licensee shall perform a liquid penetrant and, as appropriate, UT inspection of the cylinder liner landing at any time a cylinder liner is removed for any other reason. This condition is applicable to EDGs 101, 102, and 103.

5. The NRC staff will re-evalute these block top inspection requirements following the first fuel cycle.

B. Crankshafts

1. EDGs 101, 102, and 103 shall not be operated at power levels greater than 3400 KW, as indicated by the installed power meter in the Shoreham control room. If an engine is operated at a power level greater than 3400 KW, for any time, (1) the engine shall be removed from service as soon as is safely possible, (2) the engine shall be declared inoperable, and (3) the crankshaft shall be inspected in accordance with the provisions of conditions B.3 and B.4 below.

2. For the first fuel cycle, EDGs 101, 102, and 103 are each limited to two hours of cumulative operation at loads between 3300 KW and 3400 KW, in addition to the monthly surveillance tests. If this limit is exceeded, (1) the engine shall be removed from service as soon as is safely possible, (2) the engine shall be declared inoperable, and (3) the crankshaft shall be inspected in accordance with the provisions of conditions B.3 and B.4 below.

3. At each refueling outage, the licensee shall measure and record hot and cold web deflection readings on each of the diesels.

4. At the first refueling outage, the licensee shall inspect the crankpin journals numbered 5, 6, and 7, and the main bearing journals between crankpins 5, 6, and 7, including the associated oil holes and fillets in all of these journals, using LP and ET as appropriate. This condition is applicable to EDGs 101, 102, and 103.

C. Cylinder Heads

1. Cylinder heads purchased as replacements will be inspected in accordance with paragraphs (a) and (b) and subject to paragraph (c) below.

   (a) Perform an ultrasonic inspection of the firedeck of all the cylinder heads at six locations to verify that the minimum thickness requirement of .400 inch is met. The six locations are specified as follows:

      (i) The first location is on the firedeck between the exhaust gas ports approximately directly between the two exhaust gas ports.

      (ii) The second location is approximately 1 1/2" from the first location in a direction toward the exhaust side of the cylinder head.

      (iii) The third location is approximately 2" from the first location in a direction toward the intake side of the cylinder head.

      (iv) The fourth location is approximately midway between the injector port and the exhuast port on the governor side of the head.

      (v) The fifth location is approximately directly between the two intake gas ports.

      (vi) The sixth location is approximately midway between the injector and exhaust port on the flywheel side of the head.

Cylinder heads not meeting this thickness requirement shall not be used.

(b) Perform surface inspection (either liquid penetrant or magnetic particle) of intake and exhaust valve seats and the firedeck area between the exhaust valves to verify that they are free of unacceptable surface defects. Cylinder heads with unacceptable and irreparable surface defects shall not be used. Acceptance criteria are as specified in ASME III, paragraph NB-5320.

(c) Ascertain from shop records or otherwise whether any heads have through-wall weld repairs of the firedeck where the repair is performed from one side only. Any such heads shall not be used.

2. The licensee shall bar the engines over with the barring device and roll the engines over with the air start system prior to any planned starts, unless that planned start occurs within four hours of a shutdown. In addition, after engine operation, the engines shall be barred and rolled over on air after four hours but not more than eight hours after engine shutdown and then barred and rolled over once again approximately 24 hours after each shutdown. In the event an engine is removed from service for any reason other than the barring and rolling over procedure prior to expiration of the eight hour or 24 hour periods noted above, that engine need not be barred or rolled over while it is out of service. Once the engine is returned to service, the licensee shall bar the engine over and roll it over with air once at the time that it is returned to service.

3. Any head which leaks due to a crack shall be replaced.

D. Cylinder Block Cam Gallery Cracks

Cam gallery saddle locations nos. 2 and 8 on the EDG 101 and 102 blocks will be inspected by LILCO every three months, or after 30 hours of operation at or above a load of 1800 KW, whichever comes first. These inspections shall be performed from the time the EDGS are initially placed in operation for emergency standby service until the first refueling outage. Liquid penetrant examinations of the nos. 2 and 8 saddle locations shall be performed to monitor the length of the cracks, followed by TSI depth gauge measurements of the cracks to monitor their depth.

E. EDG 3300 kW Alarm

The licensee shall install a distinctive visual and audible alarm for each diesel generator in the main control room that will be set no higher than 3300 Kw for operation.

F. Procedures and Training

Prior to operation at reactor power levels greater than 5%, the licensee shall complete, in a manner acceptable to NRC staff, the development of suitable procedures and training to minimize the likelihood of operator errors that could result in EDG overload.

G. Inspection Results and Restart

If any of the engine inspections required by paragraphs A, B, and D above identify a degraded component condition, the associated engine shall be declared inoperable until the results of the inspection and further analyses by the licensee of the degraded condition have been reviewed, and further operation of the engine has been approved by the NRC staff.

ATTACHMENT 4

Changes to the Remote Shutdown Panel (RSP) (Section 7.4.3 SSER 3, SSER4)

The licensee shall provide the following additional instrumentation and controls to meet the single-failure criterion.

(1) Residual Heat Removal (RHR) system--An RHR A flow indicator will be provided on a local panel.

(2) Reactor Building Service Water (RBSW) system--Provide RBSW train A flow indication at a local panel.

(3) Spent fuel pool cooling--Provide pump controls for the A spent fuel pool cooling pump on a local panel.

(4) Miscellaneous local indicators--Provide a Division 2 indicator for Reactor Pressure Vessel (RPV) pressure and Division 1 and 2 indicators for suppression pool temperature.

The following remote shutdown panel instruments, including their sensors, power supplies, displays and associated connections shall meet the quality standards applicable to Quality Assurance Category I equipment.

(1) RHR B flow

(2) RPV level

(3) RPV pressure

(4) service water B header pressure

(5) suppression pool temperatures

(6) suppression pool level

(7) Reactor Core Isolation Cooling (RCIC) flow

(8) RCIC turbine speed

(9) SRV N_2 pressure