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MEMORANDUM FOR: Robert L. Tedesco, Assistant Director for Licensing
Division of Licensing

FROM: William V. Johnston, Assistant Director for Materials and Qualification Engineering
Division of Engineering

SUBJECT: LONG ISLAND LIGHTING COMPANY, SHOREHAM NUCLEAR POWER STATION UNIT 1

Plant Name: Shoreham Nuclear Power Station Unit 1
Suppliers: General Electric; Stone & Webster Engineering Corp.
Licensing Stage: OL
Docket Number: 50-322
Responsible Branch and Project Manager: LB-1, J. N. Wilson
Reviewer: B. J. Elliot (INEL)
Description of Task: Final Safety Evaluation Report
Review Status: Complete

The Component Integrity Section, Materials Engineering Branch, Division of Engineering has reviewed the Final Safety Analysis Report for Shoreham Nuclear Power Station Unit 1. Based on additional information supplied by the applicant in FSAR amendments through No. 39 (June 1981) and supporting documentation received via telecopier dated 6/12/81, we have revised and completed our input to the Safety Evaluation Report (SER) which is included in Attachment 1. In this SER input we have identified areas where exemptions to several requirements of Appendices G and H, 10 CFR Part 50, are required and are justified. Our evaluation of the areas that we recommend exemptions to are discussed in the attached SER sections.

/S/

William V. Johnston, Assistant Director
for Materials and Qualification Engineering
Division of Engineering

Enclosure: As stated
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ATTACHMENT 1

Long Island Lighting Company
Shoreham Nuclear Power Station Unit 1
Docket No. 50-322
Reactor Coolant Pressure Boundary Fracture Toughness

MATERIALS ENGINEERING BRANCH
COMPONENT INTEGRITY SECTION

5.3.1 Reactor Vessel Materials

General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," Appendix A, 10 CFR Part 50, requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A, 10 CFR Part 50, requires, in part, that the reactor coolant pressure boundary be designed to permit an appropriate material surveillance program for the reactor coolant pressure boundary.

The reactor vessel for Shoreham Unit 1 (Shoreham Nuclear Power Station Unit 1) was ordered in February 1967, fabricated by Combustion Engineering, subjected to General Electric's Quality Assurance Program, and was designed per the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition including Addenda through Winter 1966 based on the above purchase date. In addition, Article 4, Appendices I, II, IX, and X only of Section III, 1968 Edition, including Addenda through Winter 1968 were applied.

The Construction Permit for Shoreham Unit 1 was issued on April 14, 1973. The Edition and Addenda of the ASME Code applicable to the design and fabrication of any reactor vessel is specified in ection 50.55a of 10 CFR Part 50. Based on the reactor vessel order date and the Construction Permit date, this section of the Code of Federal Regulations requires that the Shoreham Unit 1 reactor vessel meet the requirements of at least the 1968 Edition of the ASME Code, incluc..._ \ddenda through Winter 1970. Therefore, the applicant did not comply with the explicit requirements of Paragraph 50.55a(c)(2), 10 CFR Part 50. Pursuant to Paragraph 50.55a(c)(2) of 10 CFR Part 50, we have evaluated the reactor vessel ferritic materials in accordance with the 1968 Edition of the ASME Code through Winter 1970.

1. Compliance to Appendices G and H, 10 CFR Part 50

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Requirements," of 10 CFR Part 50, specify the fracture toughness requirements for the ferritic materials of the reactor coolant pressure boundary. Because the Edition and Addenda of the ASME Code used in the design and fabrication of Shoreham Unit 1 precede the publication date of Appendices G and H, some of the fracture toughness tests were not conducted to demonstrate explicit compliance with the current requirements of Appendices G and H.

We have concluded from our review of information submitted by the applicant that exemptions to some of the specific requirements of Appendices G and H, 10 CFR Part 50, are required. Our evaluation of the areas of compliance to the requirements of Appendices G and H and the bases for granting the exemptions are discussed in the following sections of this report.

2. Evaluation of Compliance to Appendix G

Based on our review of the applicant's submittal that described the extent of compliance of Shoreham Unit 1 to Appendix G, 10 CFR Part 50, we have determined that the requirements of Appendix G have been met except for Paragraphs III.B.3, III.B.4, III.C.2, IV.A.1, IV.A.2.c, IV.A.3, and IV.B. Our evaluation of deviation from the explicit requirements of these paragraphs is contained in the following sections.

Paragraph III.B.3 of Appendix G requires that the temperature instruments and Charpy test machines be calibrated in accordance with Paragraph NB-2360 of

Section III of the ASME Code. Verification of this required calibration was impossible since the testing organization only retained the calibration report until the next calibration. However, General Electric has stated that the test instruments and machines were routinely calibrated on a periodic basis. Based on the standard practice of this period and on past experience with Charpy testing, we conclude that it is very unlikely that the test instruments and machines were not adequately calibrated and that an exemption to the requirement for maintaining the calibration report is justified.

Paragraph III.B.4 of Appendix G requires that the testing personnel shall be qualified by training and experience and should be able to perform tests in accordance with written procedures. For Shoreham Unit 1 component testing, no written procedures were in existence as required by the later regulation. However, the individuals were qualified by on-the-job training and past experience. Because these tests are relatively routine in nature and are continually being performed in the laboratory that conducted these tests, it is unlikely that the tests were conducted improperly. Consequently, we conclude that an exemption for not performing the tests in accordance with written procedures is justified.

Paragraph III.C.2 of Appendix G requires, in part, that materials used to prepare test specimens for the reactor vessel beltline region shall be taken directly from the excess material and welds in the vessel shell courses.

Paragraph III.C.2 of Appendix G was not complied with in that materials used to prepare weld test specimens for the reactor vessel beltline region were not necessarily prepared using the same base metal plate associated with the weld in the reactor vessel. The weld test specimens were taken from simulated weldments prepared from excess production plate. However, the weld wire and flux materials used in the test specimens are the same as those used in the reactor vessel beltline. After weld preparation, the weldments were subjected to a heat treatment to obtain metallurgical effects equivalent to those produced during fabrication of the reactor vessel. Based on our evaluation of this information, we conclude that although the same base materials were not used to prepare the test samples, an exemption from the specific requirements of Paragraph III.C.2 of Appendix G is justified because the same heat treatment, weld wire, flux, and welding process used in the vessel welds were used in the test specimens. Since the weld toughness properties are determined primarily by heat treatment, weld wire, flux, and welding process, and not by differences in similar base materials, the use of weldment test specimens having the same weld wire, flux, and heat treatment as the vessel welds is sufficient to satisfy the requirements of Paragraph III.C.2 of Appendix G and provides acceptable justification for an exemption to the exact requirements of Paragraph III.C.2 of Appendix G.

Paragraph IV.A.1 of Appendix G requires that a reference temperature, RT_{NDT}, be determined for each ferritic material of the reactor coolant pressure boundary and that this reference temperature be used as a basis for providing adequate margins of safety for reactor operation. The value of RT_{NDT} is defined in the ASME Code as the higher of either (a) the nil ductility temperature as determined by the dropweight test, or (b) a temperature of 60°F less than the temperature at which 50 ft-lb energy and 35 mils lateral expansion is achieved, as determined by the CVN impact test. The CVN impact test for base metal is to be conducted using specimens oriented in the transverse direction.

The value of RT_{NDT} for the base metal was determined in accordance with Paragraph IV.A.1 except that the Charpy V-notch specimens were located in the longitudinal rather than transverse direction. To compensate for specimen orientation, the temperatures at which the 50 ft-lb energy levels would have been achieved for transverse specimens were estimated as 30°F higher than the temperature indicated from the longitudinal data.

The value of RT_{NDT} for the weld metal was not in strict compliance with Paragraph IV.A.1 because the nil ductility temperature was not defined explicitly for the beltline welds and the CVN specimens were tested only at a single temperature, which in general was not sufficient to define the 50 ft-lb energy level. To define RT_{NDT} for the weld metal in accordance with Paragraph IV.A.1 the applicant employed the following alternative procedures:

a. The nil ductility transition (NDT) temperature was assumed to be -50°F.

b. If the CVN impact energy obtained at the single test temperature was less than 50 ft-lb, then the temperature at which 50 ft-lb energy would be achieved was estimated from the available data by using a temperature-impact energy correlation of 2°F per ft-lb to extrapolate the energy level obtained at the test temperature to the temperature corresponding to the 50 ft-lb energy level.

c. If the CVN impact energy obtained at the single test temperature was greater than 50 ft-lb, then the test temperature itself was used as the 50 ft-lb temperature.

We have reviewed the data obtained for Shoreham Unit 1 vessel material, the additional data supplied by the applicant for defining the NDT temperature of the welds, WRC Bulletin 217 test data, and similar test data reported for various heats of reactor pressure vessel steels in Electric Power Research Institute Reports, EPRI NP-121, Volume II, April 1976 and EPRI NP-933, December 1978. Our review of these data indicate that the correlations used by the applicant to determine the effect of specimen orientation, and the temperature at which 50 ft-lbs would be achieved is conservative and results in values of RT_{NDT} for the plate and weld materials that are equivalent to those that would be determined if the tests were conducted in strict compliance to Appendix G.

Our review of the applicant's weld NDT temperature data indicates that the assumed -50°F initial NDT temperature is conservative for all welds except for weld seams using wire from heat No. IP-2815. We estimate a conservative initial NDT temperature for those welds as -30°F. We calculated the temperature at which 50 ft-lbs would occur for those welds using method (b) above and found that it was +32°F. Thus, by NB-2330 of the ASME Code, the RT_{NDT} for this weld material would be -28°F. Therefore, in our evaluation of the reactor vessel surveillance program and pressure-temperature limits we will utilize an RT_{NDT} of -28°F for all welds fabricated using heat No. IP-2815 weld wire.

Based on the above analysis, we feel that an exemption to the requirements of Paragraph IV.A.1 of Appendix G to 10 CFR Part 50 which requires, in part, both dropweight tests and CVN impact tests on reactor beltline materials, is justified.

Paragraph IV.A.2.c of Appendix G requires, in part, that when the core is critical (other than for the purpose of low-level physics tests), the temperature of the reactor vessel shall be no less than the minimum permissible temperature for inservice system hydrostatic pressure test nor less than 40°F above the temperature required by Paragraph IV.A.2.a.

Long Island Lighting Company has requested that they be allowed to operate with the core critical at temperatures below the limits established by the inservice system hydrostatic pressure test. Shoreham FSAR in Revision No. 18 stated that the intent of the proposed alternate method of compliance with Appendix G of this vessel is to use operating limitations on pressure and temperature that provide a margin of safety against a nonductile failure of this vessel that will be equivalent to that for a vessel built to Summer 1972 Addenda of the ASME Code.

General Electric Company also has proposed that 10 CFR Part 50 be amended to make this procedure acceptable in the regulation. The proposed modification to 10 CFR Part 50, Appendix G, Paragraph IV.A.2.c is described in GE Licensing Topical Report NEDO-21778-A. As previously reported in the staff's November 13, 1978 memorandum, O. D. Parr to Dr. G. G. Sherwood, the regulatory staff has reviewed this topical report, has found it acceptable, and concurs that the proposed alternative to the criticality hydrostatic temperature limit is acceptable. Our previous evaluation and acceptance of the GE topical report provides sufficient information for the staff to conclude that the proposed alternate method is equivalent to the current Appendix G requirement, and that an exemption to Paragraph IV.A.2.c of Appendix G is justified.

Paragraph IV.A.3 of Appendix G requires that materials for piping, pumps, and valves meet the requirements of Paragraph NB-2332 of the ASME Code. All of Shoreham's RCPB valves were tested to the requirements of NB-2332 of the ASME Code except the main steam isolation valves (MSIV). The Shoreham MSIV materials were not tested because they were procured to the 1968 ASME Nuclear Pump and Valve Code, Winter 1969 Addenda, which did not require material testing.

Paragraph NB-2332 of the ASME Code requires that RCPB material used in Shoreham's MSIV be CVN impact tested at the lowest service metal temperature and that the CVN lateral expansion exceed 25 mils. Based on the lowest estimated temperature of the water in the condensate storage tank, the heat input from pumps and the temperature of the turbine building, a conservative estimate of the lowest service temperature for valves was estimated as 70°F. During MSIV operation, the lowest service metal temperature would be 212°F, because significant pressure is not applied to the MSIV until the boiling point of water, viz, 212°F.

The applicant has supplied CVN impact data for MSIV materials from several other nuclear facilities which had been fabricated to the same specification and heat treated to an equivalent metallurgical condition as the RCPB materials used in Shoreham's MSIV. The data indicate for RCPB valve materials that CVN lateral expansion at 70°F test temperature would exceed 25 mils.

An exemption to CVN impact testing MSIV RCPB materials is justified since equivalent data have been provided which demonstrates that the materials meet the CVN impact requirements of Paragraph IV.A.3 of Appendix G.

Paragraph IV.B of Appendix G requires that the reactor vessel beltline materials have a minimum upper-shelf energy, as determined from Charpy V-notch impact tests on unirradiated specimens in accordance with Paragraph NB-2322.2(a) of the ASME Code, of 75 ft-lb, unless it can be demonstrated to the Commission by appropriate data and analyses that lower values of upper-shelf energy still provide adequate margin for deterioration from irradiation.

In accordance with 10 CFR 50.55a, the fracture toughness tests were conducted to an ASME Code Edition that preceded the effective date of Appendix G to 10 CFR Part 50.. This Edition of the ASME Code did not require that the upper-shelf energy be established but only required that the tests be conducted at a single temperature equal to 60°F below the lowest service temperature. The test temperature determined in this manner typically was 10°F. However, all of the reactor vessel beltline plate material was also tested at higher temperatures.

All the reactor vessel beltline materials tested met the minimum upper-shelf requirement except for four heats of submerged arc weld metal: 20291/1092/3854, 21935/1092/3889, IP-2815/1092/3869, and 90099/0091/3458. (The numbers are heat number/flux type/flux lot number). The weld materials were tested at +10°F and all had at least one CVN impact test result less than 75 ft-lb. One had CVN impact test results less than 50 ft-lb at +10°F. No values of lateral expansion or percent shear were reported for these weld metals.

The applicant has supplied CVN impact data from other welds having the same type weld wire, flux, and heat treatment as those in the Shoreham RPV belt-line region. The CVN impact tests for these welds were performed at temperatures greater than 10°F and are a demonstration of the upper-shelf energy of the Shoreham RPV beltline welds. Since the upper-shelf energy for the demonstration welds exceeds 90 ft-lbs, we conclude that if the Shoreham RPV beltline welds had been tested at temperatures greater than 10°F, the CVN upper-shelf energy would exceed 75 ft-lbs. Based on the demonstration data submitted by the applicant, we consider that an exemption to the requirements of Paragraph IV.B that each weld be CVN impact tested to determine whether the upper-shelf energy exceeds 75 ft-lbs, is justified.

3. Evaluation of Compliance to Appendix H

Based on our review of the applicant's submittal that detailed the extent of compliance of Shoreham Unit 1 with Appendix H, 10 CFR Part 50, we have determined that the requirements of Appendix H have been met except for Paragraph II.B.

Paragraph II.B of Appendix H requires, in part, that the surveillance program for the ferritic materials in the reactor vessel beltline complies with ASTM E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessel." ASTM E 185-73 defines the type, number, and selection criteria for the reactor vessel irradiation surveillance program. The applicant is not in exact compliance with two requirements of Paragraph II.B of Appendix H. These requirements are that the limiting reactor vessel beltline materials must be included in the surveillance program and that the Charpy specimens must be oriented in the transverse direction.

The limiting materials for the Shoreham reactor vessel beltline are weld material 20291/3458 and plate material C-4803-2. The materials in the Shoreham surveillance capsule are from weld metal IP-3571/3958 and plate materials C-4882-1 and C-4882-2. Because the Shoreham Unit 1 surveillance materials are not the most limiting base plates and welds, the applicant's material surveillance program is not in full compliance with Appendix H, 10 CFR Part 50. To have an acceptable surveillance program, the applicant must recalculate the pressure-temperature limits based on the greater of the following:

(1) The actual shift in reference temperature for plate materials C-4882-1 or C-4882-2 and weld metal 20291/3458 as determined by CVN impact testing, or

(2) The predicted shift in reference temperatures for plate material C-4803-2 and weld metal 20291/3458 as determined by Regulatory Guide 1.99, Rev. 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

Although material from the most limiting weld seam and plate are not contained in the Shoreham Unit 1 materials surveillance program, we have found that an exemption to Paragraph II.B of Appendix H, 10 CFR Part 50, is justified because methods of analysis contained in Regulatory Guide 1.99, which will be used to determine the radiation induced change in fracture toughness of limiting beltline weld and plates, are conservative.

Paragraph II.B of Appendix H also requires that the CVN surveillance specimen orientation be transverse. Since the ASME Code to which the reactor vessel was built required longitudinal specimens, the applicant cannot comply with the specimen orientation requirements of Paragraph II.B. Based on our evaluation of the Shoreham Unit 1 surveillance program, we conclude that the test specimens with longitudinal orientation will provide sufficient data to predict the relative change in RT_{NDT} due to neutron irradiation. Our conclusion is based on previously obtained test data and experience that indicate that the relative shift in RT_{NDT} is not significantly sensitive to specimen orientation. Based on our evaluation, we conclude that an exemption to the specimen orientation requirements of Paragraph II.B is justified because equivalent measures of irradiation damage can be obtained from the transverse specimens.

Data submitted by the applicant indicate that the highest adjusted RT_{NDT} is 169°F. Therefore, according to Paragraph II.C.3.b of Appendix H, four surveillance capsules are necessary. Also, the withdrawal of the first surveillance capsule for Shoreham Unit 1 must occur within two effective full power years (EFPY).

The applicant has placed only three capsules in the reactor vessel and plans to add a fourth surveillance capsule after removal of the first capsule. The applicant indicated that the removal of the first capsule would occur at the end of the first 10-year interval. To conform to the withdrawal sequence of Paragraph II.C.3.b of Appendix H, 10 CFR Part 50, the applicant's technical specification for Shoreham Unit 1 must indicate that the surveillance capsules must be withdrawn during the refueling outage which occurs prior to the following effective full power years.

Surveillance Capsule 1: 2 effective full power years
Surveillance Capsule 2: 13 effective full power years
Surveillance Capsule 3: 24 effective full power years
Surveillance Capsule 4: Standby

4. Conclusions for Compliance to Appendices G and H, 10 CFR Part 50

Our technical evaluation has not identified any practical methods by which the existing Shoreham Unit 1 reactor vessel can comply with the specific requirements of Paragraphs III.B.3, III.B.4, III.C.2, IV.A.1, IV.A.2.c, IV.A.3, and IV.B of Appendix G and Paragraph II.B of Appendix H, 10 CFR Part 50. However, the alternate methods proposed to demonstrate compliance with these paragraphs of Appendices G and H have been reviewed and evaluated, and have been found to demonstrate that the safety margins required by Appendices G and H have been achieved.

Based on the foregoing, pursuant to 10 CFR 50.12, exemptions from the specific requirements of Appendices G and H of 10 CFR Part 50, as discussed above, are authorized by law and can be granted without endangering life or property or the common defense and security and are otherwise in the public interest. We conclude that the public is served by not imposing certain provisions of Appendices G and H of 10 CFR Part 50 that have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

Furthermore, we have determined that the granting of these exemptions does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. We have concluded that these exemptions would be insignificant from the standpoint of environmental impact and, pursuant to 10 CFR 51.5(d)(4), an environmental impact statement or negative declaration and environmental appraisal need not be granted in connection with this action.

5.3.2 Pressure-Temperature Limits

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Materials Surveillance Program Requirements," 10 CFR Part 50, describe the conditions that require pressure-temperature limits for the reactor coolant pressure boundary and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins for the reactor coolant pressure boundary at least as great as the safety margins recommended in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure." Appendix G, 10 CFR Part 50, requires additional safety margins whenever the reactor core is critical, except for low-level physics tests.

The following pressure-temperature limits imposed on the reactor coolant pressure boundary during operation and tests are reviewed to ensure that they provide adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components as required by General Design Criterion 31:

(1) Preservice hydrostatic tests,

(2) Inservice leak and hydrostatic tests,

(3) Heatup and cooldown operations, and

(4) Core operation.

Appendices G and H, 10 CFR Part 50, require the applicant to predict the shift in reference temperature due to neutron irradiation. The shift in RT_{NDT} due to neutron irradiation is then added to the initial RT_{NDT} to establish the adjusted reference temperature. The base plate or weld seam having the highest adjusted reference temperatures is considered the most limiting materials upon which the pressure-temperature operating limits are based. In the case of Shoreham Unit 1 our estimate of the most limiting material using the methods contained in Regulatory Guide 1.99, Rev. 1, is plate C-4803-2 for the first 10 effective full power years (EFPY) and weld material 20291/3458 for the period from 10 EFPY until end of plant life. Once in service, the pressure-temperature limits must be revised to reflect the actual neutron radiation damage as determined from the results of the reactor vessel materials surveillance program.

According to our evaluation the proposed heatup and cooldown pressure-temperature limits are acceptable for 1.6 EFPY. During the refueling which is prior to 1.6 EFPY, the applicant will verify the predicted neutron fluence by dosimetry measurements. This dosimetry measurement will then be utilized to predict the neutron fluence for calculating the pressure temperature limit curves subsequent to fuel reloading. The calculated shift in RT_{NDT} for the reactor vessel beltline must be based on Regulatory Guide 1.99. After removal of the first surveillance capsule the applicant must recalculate the pressure temperature limit curves based on the analysis discussed in SER Section 5.3.1.

The pressure-temperature limits to be imposed on the reactor coolant system for all normal operating, testing, and anticipated transient conditions, to ensure adequate safety margins against nonductile or rapidly propagating failure, are in conformance with established criteria, codes, and standards acceptable to the staff. The use of operating limits based on these criteria, as defined by applicable regulations, codes, and standards, provides reasonable assurance that nonductile or rapidly propagating failure will not occur, and constitutes an acceptable basis for satisfying the applicable requirements of General Design Criterion 31.

5.3.3 Reactor Vessel Integrity

We have reviewed the FSAR sections related to the reactor vessel integrity of Shoreham Unit 1. Although most areas are reviewed separately in accordance with other review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted.

We have reviewed the information in each area to ensure that it is complete and that no inconsistencies exist that would reduce the certainty of vessel integrity. The areas reviewed are:

1. Design (SER §5.3.1)

2. Materials of construction (SER §5.3.1)

3. Fabrication methods (SER §5.3.1)

4. Operating conditions (SER §5.3.2)

We have reviewed the above factors contributing to the structural integrity of the reactor vessel and conclude that the applicant has complied with Appendices G and H, 10 CFR Part 50, except for Paragraphs III.B.3, III.B.4, III.C.2, IV.A.1, IV.A.2.c, and IV.A.3, and IV.B of Appendix G, and Paragraph II.B of Appendix H, for which the applicant has provided sufficient information to justify exemptions.

Paragraph III.B.3 of Appendix G requires that the temperature instruments and Charpy test machines be calibrated per Paragraph NB-2360 of the ASME Code. The standard practice of the time and past experience with Charpy testing make it unlikely that the test instruments were not adequately calibrated and that an exemption to Paragraph III.B.3 is justified.

Paragraph III.B.4 of Appendix G requires the applicant to qualify the personnel performing the CVN impact testing according to written procedures. Although the personnel were not qualified by written procedures, the individuals conducting the CVN impact tests were qualified by on-the-job training and past experience which we conclude is adequate and sufficient to justify an exemption to Paragraph III.B.4 of Appendix G.

Paragraph III.C.2 of Appendix G requires that the base metal used to prepare test specimens be taken from excess base metal from the vessel beltline region. The weld specimens for testing were not prepared from excess production plate. The applicant, however, has supplied sufficient data to demonstrate that the weld specimens do represent the welds in the vessel beltline region. Therefore, an exemption to Paragraph III.C.2 is justified.

Paragraph IV.A.1 of Appendix G requires that a reference temperature, RT_{NDT}, be determined per Paragraph NB-2330 of the ASME Code for each ferritic material in the reactor coolant pressure boundary. Although the applicant did not determine the RT_{NDT} per Paragraph NB-2330 of the ASME Code for each ferritic material, the critical RT_{NDT} for operating, maintenance, and testing conditions has been determined based on additional information available in the literature and additional data supplied by the applicant. Therefore, we have concluded that an exemption to paragraph IV.A.1 of Appendix G is justified.

Paragraph IV.A.2.c of Appendix G requires, in part, that when the core is critical (other than for low-level physics tests), the temperature of the reactor vessel shall be no less than the minimum permissible temperature for inservice system hydrostatic pressure test nor less than 40°F above the temperature required by Paragraph IV.A.2.a. Long Island Lighting has requested that they be allowed to operate with the core critical at temperatures below the limits established by the inservice system hydrostatic pressure test. Based on the memorandum dated November 13, 1978, O. D. Parr to Dr. G. G. Sherwood in which the staff concurred that the proposed alternative to criticality hydrostatic temperature limit is acceptable, we consider an exemption to Paragraph IV.A.2.c, Appendix G, is justified.

Paragraph IV.A.3 of Appendix G requires, in part, that the materials for valves meet CVN impact requirements in Paragraph NB-2330 of the ASME Code. Although the applicant has not CVN impact tested the MSIV materials, the applicant has supplied sufficient data from other similar materials to demonstrate that the MSIV would meet the CVN impact requirements of Paragraph NB 2330 of the ASME Code and, therefore, an exemption to Paragraph IV.A.3 is justified.

Paragraph IV.B, Appendix G, requires that the reactor vessel beltline materials have a minimum upper-shelf CVN energy of 75 ft-lb unless it can be demonstrated that lower values of upper-shelf CVN energy still provide adequate margin for irradiation deterioration. Although the applicant has not tested all reactor vessel beltline material over a sufficient temperature range to determine whether each material has a minimum upper-shelf energy of 75 ft lbs, the applicant has supplied sufficient information from other plants to demonstrate that the CVN impact under-shelf energies for the Shoreham Unit 1 reactor vessel beltline materials exceed 75 ft lbs. Therefore, we conclude that an exemption to Paragraph IV.B is justified.

Paragraph II.B, Appendix H, requires that the material surveillance program comply with ASTM E 185-73. The materials in Shoreham Unit 1's surveillance program does not comply with all requirements in ASTM E 185; however, the materials that are in the program, together with methods for predicting radiation damage, provide sufficient information for us to have concluded that an exemption to Paragraph II.B, Appendix H, is justified.

We have reviewed all factors contributing to the structural integrity of the reactor vessel and conclude there are no special considerations that make it necessary to consider potential reactor vessel failure for Shoreham Unit 1.

LILCO

LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION
P O BOX 618 NORTH COUNTRY ROAD • WADING RIVER N Y 11792

August 26, 1981 SNRC-616

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Shoreham Nuclear Power Station-Unit 1
Docket No. 50-322

Dear Mr. Denton:

Enclosed herewith are fifteen (15) copies of a document prepared by the Long Island Lighting Company entitled "Compliance of Shoreham Nuclear Power Station - Unit 1 with the NRC Regulations of 10CFR Parts 20, 50, and 100". This document addresses those sections of the regulations which impose requirements on Shoreham.

If you require additional information, please do not hesitate to contact this office.

Very truly yours,

B.R. McCaffrey
B. R. McCaffrey
Manager, Project Engineering
Shoreham Nuclear Power Station

cc/pg

Enc.

cc: J. Higgins

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Summary of Compliance

with the

Code of Federal Regulations

Title 10 - Energy

Chapter 1 - Nuclear Regulatory Commission

Parts 20, 50 and 100

January 1, 1981

for

Shoreham Nuclear Power Station - Unit 1

Long Island Lighting Company

Introduction

This document presents a summary of the compliance of the Shoreham Nuclear Power Station - Unit 1 (SNPS) design and operation to the NRC regulations put forth in 10 CFR Parts 20, 50 and 100. Those sections of the regulations which are applicable to Shoreham and impose requirements on applicants for and holders of Operating Licenses are addressed. Regulations pertaining to Construction Permits are not discussed.

ENCLOSURE
COMPLIANCE OF SHOREHAM NUCLEAR POWER STATION UNIT 1
WITH THE NRC REGULATIONS OF 10CFR PARTS 20, 50, and 100

Regulation (10CFR)	Compliance
20.1 (c)	Conformance to the ALARA principle stated in this regulation is ensured by implementation of the management policy stated in Section 12.1.1.1 of the FSAR. This implementation encompasses appropriate Technical Specifications, Health Physics procedures, delegation of responsibility and an on-going training program. Chapters 11 and 12 of the FSAR describe the specific equipment and design features utilized in this effort.
20.3	The definitions contained in this regulation are adhered to in applicable sections of the FSAR and all appropriate Technical Specifications and procedures.
20.4	The Units of Radiation Dose specified in this regulation are all applicable SNPS procedures.
20.5	The Units of Radioactivity specified in this regulation are used in all applicable SNPS procedures.
20.101	The radiation dose limits specified in this regulation are complied with through the implementation of and adherence to administrative policies and controls, and in appropriate Health Physics procedures developed for this purpose. Conformance is documented by the use of appropriate personnel monitoring devices and the maintenance of all required records (See FSAR Chapter 12).
20.102	When required by this regulation, before permitting any individual to exceed the exposure limits specified in 20.101(a) previous accumulated occupational dose is determined by the use of the equivalent to

Regulation (10 CFR)	Compliance
20.102 (con't.)	Form NRC-4. Appropriate Health Physics procedures and administrative policies control this process. (See FSAR Chapter 12).
20.103 (a)	Compliance with this regulation is ensured through the implementation of appropriate Health Physics procedures relating to air sampling for radioactive materials, and bioassay of individuals to measure radioactivity Administrative policies and controls will provide adequate margins of safety for the protection of individuals to measure radioactivity in the body. Administrative policies and controls will provide adequate margins of safety for the protection of individuals against the intake of radioactive materials. The systems and equipment described in Chapters 11 and 12 of the FSAR provide the capability to minimize these hazards.
20.103 (b)(1)	The reactor building ventilation system is designed to provide a means to reduce the concentration of particulate and gaseous contamination to assure safe continuous access (40 hours/week) during normal reactor shutdown (See FSAR Section 12.3.3) Portable ventilation systems, hoods, and tents are used as practicable to contain and reduce airborne particulate and gaseous contamination during the performance of various jobs.
20.103 (b)(2)	Compliance with this regulation is ensured through the implementation of appropriate Health Physics procedures relating to air sampling for radioactive materials, and bioassay of individuals for internal contamination. Administrative policies and controls will provide adequate margins of safety for the protection of individuals against the intake of radioactive materials. The systems and equipment described in Chapters 11 and 12 of the FSAR provide the capability to minimize these hazards.

Regulation (10 CFR)	Compliance
20.103 (b) (2) (Con't.)	Issuance and Selection of Respiratory Equipment Health Physics personnel at SNPS will select appropriate respiratory equipment so that contaminant concentration inhaled by the wearer does not exceed the appropriate regulatory limits specified in Appendix B, Table I, Column 1. Should an individual receive greater than 40 Maximum Permissible Concentration (MPC) hours in 7 consecutive days, an evaluation will be made to identify the cause and actions will be taken to prevent recurrence. Records will be maintained for each occurrence.
20.103 (c)	Health Physics personnel at SNPS will select appropriate respiratory equipment so that contaminant concentration inhaled by the wearer does not exceed the appropriate regulatory limits. The protection factors used at SNPS comply with the protection factors permitted under Regulatory Guide 8.15.
20.103 (e)	The NRC Region I Director shall be notified at least 30 days before respiratory protective equipment is first used under the provisions of this section.
20.104	Conformance with this regulation is accomplished through the use of the appropriate Health Physics procedures. Access to restricted areas is minimized to assure that minors do not receive a dose in excess of 10 percent of the limits specified in 20.101, paragraph (a).
20.105 (a)	This regulation allows information concerning anticipated average radiation levels and anticipated occupation times for each of the unrestricted areas to be submitted. Chapters 11 and 12 of the FSAR provide the information and related dose assessments. LILCO has not and does not currently intend to modify limits for unrestricted areas.

Regulation (10 CFR)	Compliance
20.105 (b)	The radiation dose rate limits specified in this regulation will be complied with through the implementation of SNPS procedures, Technical Specifications, and administrative policies which control the use and transfer of radioactive materials. Appropriate surveys and monitoring devices will document this compliance.
20.106 (a)	Conformance with the limits specified in this regulation will be assured through the implementation of SNPS procedures and applicable Technical Specifications which provide adequate sampling and analyses, and monitoring of radioactive materials in effluents before and during their release. The level of radioactivity in station effluents will be minimized to the extent practicable by the use of appropriate equipment designed for this purpose, as described in Chapter 11 of the FSAR.
20.106 (b)&(c)	LILCO has not and does not currently intend to include in any license or amendment applications proposed limits higher than those specified in 20.106(a), as provided in these regulations.
20.106 (d)	Appropriate allowances for dilution, dispersion and decay of radioactive effluents will be made in conformance with this regulation, and are described in detail in Chapter 11 of the FSAR.
20.108	Necessary bioassay equipment and procedures, including Whole Body Counting, will be utilized to determine exposure of individuals to concentrations of radioactive materials. Appropriate Health Physics procedures and administrative policies implement this requirement. (See FSAR Chapter 12.)
20.201	The surveys required by this regulation will be performed at adequate frequencies and contain such detail as to be consistent with the radiation hazard being evaluated.

Regulation (10 CFR)	Compliance
20.201 (Con't.)	When necessary, the Radiation Work Permit program established at the station provides for detailed physical surveys of equipment, structures and work sites to determine appropriate levels of radiation protection. The SNPS Administrative Procedures and applicable Health Physics procedures will require these surveys and provide for their documentation in such manner as to ensure compliance with the regulations of 10CFR Part 20. (See FSAR Chapter 12.)
20.202 (a)	The applicable SNPS Health Physics procedures will set forth policies and practices which ensure that all individuals are supplied with, and required to use, appropriate personnel monitoring equipment. The Radiation Work Permit system will be established to provide additional control of personnel working in radiation areas and to ensure that the level of protection afforded to these individuals is consistent with the radiological hazards in the work place. (See FSAR Chapter 12.)
20.202 (b)	The terminology set forth in this regulation will be used in all applicable SNPS Health Physics procedures, Technical Specifications, and Administrative Procedures.
20.203 (a)	All materials used for labeling, posting, or otherwise designating radiation hazards or radioactive materials, and using the radiation symbol, conform to the conventional design prescribed in this regulation per SNPS Health Physics procedures.
20.203 (b)	This regulation will be conformed to through the implementation of appropriate Health Physics procedures relating to posting of radiation areas, as defined in 10 CFR Part 20.202 (b)(2).
20.203 (c)	The requirements of this regulation for "High Radiation Areas" will be conformed to by the implementation of the SNPS Technical Specifications and appropriate SNPS Health

Regulation (10 CFR)	Compliance
20.203 (c) (Con't.)	Physics procedures. The controls and other protective measures set forth in the regulation are maintained under the surveillance of the SNPS Health Physics section.
20.203 (d)	Each Airborne Radioactivity Area, as defined in this regulation, will be required to be posted by provisions of the appropriate SNPS Health Physics procedures. These procedures also provide for the surveillance requirements necessary to determine airborne radioactivity levels.
20.203 (e)	The area and room posting requirements set forth in this regulation pertaining to radioactive materials will be complied with through the implementation of appropriate SNPS Health Physics procedures.
20.203 (f)	The container labeling requirements set forth in this regulation will be complied with through the implementation of appropriate Health Physics procedures.
20.204 (a),(c),(d)	The posting requirement exceptions described in this regulation are used where appropriate and necessary at SNPS. Adequate controls are provided within the SNPS health physics procedures to ensure safe and proper application of these exceptions.
20.205	The requirements of this regulation pertaining to procedures for picking up, receiving, and opening packages of radioactive materials will be implemented by the appropriate SNPS Health Physics procedures. These procedures also provide for the necessary documentation to ensure an auditable record of compliance.
20.206	The requirements of 10 CFR 19.12 referred to by this regulation are satisfied by the radiation worker training conducted at SNPS. Appropriate administrative procedures set forth requirements for all radiation workers to receive this instruction on a periodic basis. (See FSAR Chapter 12.)

Regulation (10 CFR)	Compliance
20.207	Licensed materials at SNPS are controlled through the Licensed Source Users Committee. If licensed materials are in unrestricted areas they will be stored and/or controlled in accordance with this section.
20.301	The general requirements for waste disposal set forth in this regulation are complied with through SNPS Health Physics procedures, the Technical Specifications, and the provisions of the station operating license. Chapter 11 of the FSAR describes the Solid Waste Disposal System installed at SNPS.
20.302	No such application for proposed disposal procedures, as described in this regulation, has been made or is currently contemplated at LILCO.
20.303	No plans for waste disposal by release into sanitary sewerage systems, as provided for in this regulation, are contemplated by SNPS, nor is this practice currently utilized.
20.304	No plans for waste disposal by burial in soil, as provided for in this regulation are contemplated by SNPS, nor is this practice currently utilized.
20.305	Specific authorization, as described in this regulation, is not currently being sought by LILCO for treatment or disposal of wastes by incineration.
20.401	All of the requirements of this regulation will be complied with through the implementation of appropriate Technical Specifications and health physics procedures pertaining to records of surveys, radiation monitoring and waste disposal. The retention periods specified for such records are also provided for in these specifications and procedures. (See FSAR Chapter 12.)
20.402	SNPS has established an appropriate inventory and control program to ensure strict accountability for all licensed radioactive

Regulation (10 CFR)	Compliance
20.402 (Con't.)	materials. Reports of theft or loss of licensed material will be required by reference to the applicable regulations and in the Technical Specifications.
20.403	Notifications of incidents, as described in this regulation, will be assured by the requirements of the Technical Specifications, the SNPS Administrative Procedures and appropriate plant procedures, which also provide for the necessary assessments to determine the occurrence of such incidents.
20.405	Reports of overexposures to radiation and the occurrence of excessive levels and concentrations, as required by this regulation, will be provided for in the Technical Specifications and in appropriate Health Physics procedures.
20.407	The requirements of personnel monitoring reporting will be complied with and appropriate Health Physics procedures established to generate a data base.
20.408	The report of radiation exposure required by this regulation upon termination of an individual's employment or work assignment is generated through the provisions of the Health Physics procedures.
20.409	The notification and reporting requirements of this regulation, and those referred to by it, are satisfied by the provisions of the Health Physics procedures.
50.10	An Operating License has been applied for in accordance with applicable regulations and is currently under review by the Commission.
50.30	This regulation sets forth procedural requirements for the filing of license applications. LILCO has complied with the procedural requirements in effect at the time of filing its license application and amendments thereto.

Regulation (10 CFR)	Compliance
50.33	This regulation requires the license application to contain certain general information such as identification of applicant, type of license sought, financial qualifications of the applicant, scheduled completion date, and a list of regulatory agencies with jurisdiction over the applicant's rates and services. This information is provided in the Shoreham Operating License Application.
50.34 (b)	The Final Safety Analysis Report (FSAR) was initially submitted on August 28, 1975, and was subsequently docketed pursuant to 10 CFR 50.34. The FSAR contains the information required by this regulation. The document, along with its numerous amendments, has been and is undergoing extensive review by the NRC and its staff.
50.34 (c)	The Physical Security Plan for SNPS was submitted to the Nuclear Regulatory Commission (NRC) in October 1977 with the most recent amendment (Rev. 4) submitted April 6, 1981. The NRC has completed its review of the Plan.
50.34 (d)	The Safeguards Contingency Plan for SNPS was submitted to the NRC on 3/23/79. This plan was most recently amended on 6/10/81 (Rev. 2). The NRC has completed its review of the Plan.
50.34a (c)	The required information pursuant to 50.34a is presented in Chapter 11 of the FSAR. The Environmental Report-Operating License stage provides the required information on expected releases.
50.36 (a)&(b)	The Shoreham application for an Operating License incorporates Technical Specifications in accordance with the requirements of 10 CFR 50.36. Revisions to these Technical Specifications are presently being prepared.
50.36 (c)	This regulation lists the categories which will be included in the Technical Specifications, such as safety limits, limiting

Regulation (10 CFR)	Compliance
50.36 (c) (Con't.)	conditions for operation, surveillance requirements, design features and administrative controls. All of the above listed categories are contained in the current draft Technical Specifications which are under preparation.
50.36a	SNPS "Compliance with 10 CFR 50 Appendix I", Docket 50-322, July 1976, demonstrates the design capability established to keep releases as low as reasonably achievable. The SNPS Radiological Effluent Technical Specification will ensure that concentrations of radioactive effluents released to unrestricted areas are within the limits specified in 10 CFR 20.106 and that the semiannual effluent report requirements are met.
50.37	This regulation requires the applicant to agree to limit access to Restricted Data. As stated in the Operating License application ". . . LILCO hereby agrees that it will not permit any individual to have access to Restricted Data until the Civil Service Commission shall have made an investigation and report to the Commission on the character, associations, and loyalty of such individual and the Commission shall have determined that permitting such person to have access to Restricted Data will not endanger the common defense and security". (paragraph II. (j))
50.38	This regulation prohibits the NRC from issuing a license to foreign-controlled entities. LILCO's statement that it is not owned, controlled, or dominated by an alien, foreign corporation, or foreign government is in the Operating License application for the Shoreham Nuclear Power Station. (see paragraph II. (d)(3)(iii))
50.40 (a)	The design and operation of the facility is to provide reasonable assurance that the Applicant will comply with NRC regulations, including those in 10 CFR Part 20, and that the health and safety of the public will not be endangered.

Regulation (10 CFR)	Compliance
50.40 (a) (Con't.)	The basis for LILCO's assurance that the regulations will be met and the public protected is contained in this enclosure and in the License Application and the related correspondence over the years. Moreover, the lengthy process by which the plant is designed, constructed, and reviewed, including reviews by LILCO's staff, G.E. and S&W staffs, the NRC staff, the ACRS, and NRC Licensing Boards, provides a great deal of assurance that the public health and safety will not be endangered.
50.40 (b)	The Atomic Safety and Licensing Board at the Construction Permit stage found LILCO to be technically and financially qualified to design and construct Shoreham.
50.40 (c)	Another consideration is that the issuance of the license is not to be inimical to the common defense and security or to the health and safety of the public. The individual showings of compliance with particular regulations contained in this enclosure, as well as the contents of the entire FSAR and related correspondence over the years, plus the lengthy process of design, construction, and review by LILCO's staff, G.E. and S&W staffs, the NRC Staff, the ACRS, and NRC Licensing Boards provide considerable assurance that the license will not be inimical to the health and safety of the public. There is considerable assurance that the license will not be inimical to the common defense and security in that LILCO has an approved security plan for Shoreham Nuclear Power Station, is not controlled by agents of foreign countries, and has agreed to limit access to Restricted Data.
50.40 (d)	The final consideration of 50.40 is that applicable requirements of Part 51 have satisfied. Part 51 concerns compliance with the National Environmental Policy Act of 1969. LILCO submitted a Final Environmental Report. The NRC reviewed the report and published a final Environment Impact

Regulation (10 CFR)	Compliance
50.40 (d) (Con't.)	Statement, NUREG-0285, in October 1977, pursuant to 10 CFR 51. Additionally, the ASLB has closed the environmental phase of the Operating License hearings.
50.42	Section 50.42 provides additional "considerations" to "guide" the Commission in issuing Class 103 licenses. The two considerations are: (a) that the proposed activities will serve a useful purpose proportionate to the quantities of special nuclear material or source material to be utilized and (b) that due account will be taken of the antitrust advice provided by the Attorney General under subsection 105c of the Atomic Energy Act. The "useful purpose" to be served is the production of electric power. The need for the power was determined by the Licensing Board at the Construction Permit stage. Although conditions affecting the need for power are constantly changing, LILCO periodically makes load projections and, in LILCO's judgment, the need for Shoreham Nuclear Station is still substantial. As for the amount of special nuclear material or source material used, there is no reason to believe that their proportion in relation to the power produced is substantially greater than that of other commercial power reactors in this country. Updated antitrust information was submitted via LILCO letter SNRC-509 from J.P. Novarro to H.R. Denton, dated September 30, 1980. The NRC is presently undertaking the antitrust review and has not yet informed the Applicant of its conclusions.
50.43 (c)	Long Island Lighting Company, which transmits electric energy in interstate commerce, and sells it at wholesale in interstate commerce, is in compliance with the regulatory provisions of the Federal Power Act.

Regulation (10 CFR)	Compliance
50.44	Compliance with paragraphs (a) through (d) and (f) is discussed in FSAR Section 6.2.5. The primary containment will be provided with an inerted atmosphere as discussed in the response to NUREG-0737, Item II.B.7 (See SNRC-608, dated July 31, 1981). Paragraph (e) is not applicable, since the notice of hearing on the Application for the Construction Permit for Shoreham was published prior to November 5, 1970. Paragraph (g) is not applicable, since the notice of hearing on the Application for the Construction Permit for Shoreham was published after December 22, 1968. As stated the combustible gas control systems for Shoreham meet the requirements of 10 CFR 50.44.
50.46	Compliance with 10 CFR 50.46 is documented in the FSAR, Section 6.3.3. This analysis shows that SNPS meets 10 CFR 50.46 criteria and the ECCS equipment will perform its function in an acceptable manner.
50.48	LILCO's compliance with this regulation is as stated in this enclosure where compliance with Appendix R is discussed.
50.51	In the Operating License application, LILCO requested that the license be issued for a period of 40 years.
50.53	This regulation provides that licenses are not to be issued for activities that are not under or within the jurisdiction of the United States. The operation of Shoreham Nuclear Power Station will be within the United States and subject to the jurisdiction of the United States, as is evident from the description of the facility in the Operating License application.

Regulation (10 CFR)	Compliance
50.54	This regulation specifies certain conditions that are incorporated in every license issued. Compliance is effected simply by including these conditions in the license when it is issued. Indeed, much of 50.54 merely provides that other provisions of the law apply, which would be the case even without 50.54. LILCO will comply with these requirements on the timetable set in the regulation.
50.55a (a)(1)	Various chapters of the FSAR discuss design, fabrication, erection, construction, testing, and inspection of safety-related equipment. For example, Section 5.2 discusses the design of the reactor coolant system, Chapter 14 covers testing of various safety-related systems, and Chapter 17 gives information on the quality assurance program utilized in the inspection of equipment.
50.55a (a)(2)	This subparagraph provides general information leading into subparagraphs (c) through (i) of this regulation.
50.55a (b) (1)&(2)	These subparagraphs provide guidance to be used in applying subparagraphs (c) through (i) concerning the approved edition and Addenda of Sections III and XI of the ASME Boiler and Pressure Vessel Code.
50.55a (c)	Design and fabrication of the reactor vessel was carried out in accordance with ASME Section (II (1965) Class A including Addenda through winter (1966). Information can be found in Chapters 3 and 5 of the FSAR.
50.55a (d)	Reactor coolant system piping meets the requirements of ASME Section III or ANSI B.31.1. Chapters 3 and 5 of the FSAR contain further information and specifically, Table 3.2.1.1 outlines the applicable design codes and the associated purchase order dates.
50.55a (e)	Pumps which are part of the reactor coolant pressure boundary meet the requirements of

Regulation (10 CFR)	Compliance
50.55a (e) (Con't.)	the ANSI B31.1. Information can be found in Chapters 2 and 5 of the FSAR.
50.55a (f)	The valves which are part of the reactor coolant pressure boundary were designed and fabricated in accordance with the requirements of ASME Section III, or ANSI-B31.1. (See Chapters 2 and 5 of the FSAR)
50.55a (g)	Inservice inspection requirements for the Shoreham plant are to be stated in Section 4.0.5 of the Shoreham Technical Specifications and reflect the requirements of 10 CFR 50.55 a(g). The Shoreham ISI program is being developed based on the 1977 edition through summer 1978 addenda of the ASME B&PV Code Section XI. In addition, a pump and valve operability program has been implemented in accordance with NRC staff guidance and sections IWP and IWV of Section XI of the code.
50.55a (h)	As discussed in FSAR Chapter 7, the protection systems meet IEEE 279-1971.
50.55a (i)	Fracture toughness requirements are set forth in Appendices G and H of 10 CFR 50. Compliance with Appendices G and H is outlined in detail in FSAR Tables 5.2.4-1 and 2, respectively.
50.58	This regulation provides for the review and report of the Advisory Committee on Reactor Safeguards. A review by the ACRS will be scheduled.
50.59	As discussed in FSAR section 13.4.2.2, the LILCO Nuclear Review Board (NRB) shall review: a) The safety evaluation completed under the provision of 10 CFR 50.59 for proposed 1) changes to procedures, equipment, or systems, and 2) tests or experiments to verify that such actions do not constitute an unreviewed safety question; b) Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59; and

Regulation (10 CFR)	Compliance
50.59 (Con't.)	c) Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.
	The NRB shall report to and advise the Vice President-Nuclear of their activities. Safety evaluations of changes made to the equipment or review of tests and experiments to comply with 10 CFR 50.59 shall be kept in a manner convenient for review and shall be retained for at least five years. SNPS will comply with the reporting requirements set forth in 10 CFR 50.59.
50.70	The Commission has assigned a resident inspector to the SNPS. LILCO has provided office space in accordance with the requirements of this section. LILCO permits access to the Station to NRC inspectors in accordance with 10 CFR 50.70 (b)(3).
50.71	Records are and will be maintained in accordance with the requirements of Paragraphs (a) through (e) of this regulation and license. Paragraph (e) requires that the FSAR be updated (within 24 months of the issuance of the operating license and annually thereafter). Such updates will be made in accordance with this Section.
50.72	Notification of significant events to the NRC will be made in accordance with this regulation. Initiation of the Emergency Plan will result in notification to the NRC Operations Center. Any other reportable occurrence which may result during normal plant operation will be reported as specified in the Shoreham Technical Specification.
50.90	LILCO will comply with the provisions of this section when applying for an amendment of Shoreham's Operating License.
Appendix A	Compliance with the General Design Criteria of this Appendix is discussed in FSAR Section 3.1. Furthermore, compliance with General Design Criteria is demonstrated throughout the FSAR and other materials which have been submitted on the docket to the NRC.

Regulation (10 CFR)	Compliance
Appendix B	Chapter 17 of the FSAR describes the provisions of the Quality Assurance Program which has been implemented to comply with the applicable requirements of this Appendix.
Appendix C	This regulation describes the information which is required to establish financial qualifications. The Atomic Safety and Licensing Board at the Construction Permit stage found LILCO to be financially qualified to design and construct Shoreham. The NRC is currently reviewing the latest information on financial qualifications by LILCO.
Appendix E	This regulation sets forth the criteria for emergency planning. LILCO submitted its emergency plan for Shoreham on May 27, 1981 via SNRC-568. The NRC has not yet informed the applicant of the conclusions of their review.
Appendices G, H	Compliance with Appendices G and H has been assessed in FSAR Section 5.2.4 and the response to SER open item #20 transmitted to the NRC in LILCO letter SNRC-578 dated 5/29/81.
Appendix I	This Appendix provides numerical guides for design objectives and limiting conditions for operation to meet the criteria "as low as is reasonably achievable" for radioactive material in light-water-cooled nuclear power reactor effluents. LILCO has filed with the Commission the necessary information to demonstrate compliance and permit an evaluation of the Shoreham Nuclear Power Station with respect to the requirements of Appendix I, as documented in "Shoreham Nuclear Power Station - Unit 1, Compliance with 10 CFR 50 Appendix I" transmitted to the NRC via SNPS-119, dated July 30, 1976.

Regulation (10 CFR)	Compliance
Appendix J	Reactor pressure boundary leakage testing for water cooled power reactors is delineated in this Appendix. Compliance with these requirements is outlined in Technical Specification 3/4.6.1 and FSAR Section 6.2.1.4.1. The special testing requirements of the Main Steam Isolation Valves are addressed in FSAR Section 3.6.1.2.
Appendix K	Compliance with 10 CFR 50 Appendix K is specifically required by 10 CFR 50.46. Compliance is documented in Section 6.3.3 of the SNPS FSAR.
Appendix R	This appendix delineates general and specific requirements concerning a fire protection program and sets forth certain fire protection features required to satisfy Criteria 3 of Appendix A to 10 CFR 50. Shoreham's compliance to this Appendix is addressed in letter SNRC-572 dated May 21, 1981.
100.10	This regulation sets forth factors to be considered when evaluating a site, such as characteristics of reactor design and operation, population density and use characteristics of the site environs, and physical characteristics of the site. All of the above factors have been provided in the application. Site physical characteristics, including seismology, meteorology, geology and hydrology, and population density and site use characteristics, including the exclusion area, low population zone, and population center distance, are presented in Chapter 2 of the FSAR. The FSAR also describes the characteristics of reactor design and operation.
100.11	This regulation sets forth the means to derive an exclusion area, a low population zone, and population center distance. All the requirements of this regulation with regard to the above distances and area are met and are described in Chapter 2 of the FSAR.

Regulation (10 CFR)	Compliance
Appendix A	Structures and equipment important to plant safety are protected from or designed to withstand all appropriate seismically related phenomena at the plant site. Design is based on the most severe probable seismic event with special consideration for the uncertainty in prediction. Detailed discussions of the development of the design criteria and how they are applied to the structures and equipment, are found in the following FSAR sections: Geology and Seismology-Section 2.5 Classification of Structures, Systems, and Components-Section 3.2 Water Level (Flood) Design-Section 3.4 Seismic Design-Section 3.7 Design of Category I Structures-Section 3.8 Mechanical Systems and Components-Section 3.9, Seismic Design of Category I Instrumentation and Electrical Equipment-Section 3.10.