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SHOREHAM NUCLEAR POWER STATION
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JOHN D. LEONARD, JR.
VICE PRESIDENT - NUCLEAR OPERATIONS

SNRC-1577

APR 5 1989

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Completion Status of License Conditions
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Gentlemen:

The following information reaffirms previous commitments and updates the completion status of the license conditions which appear in Section 2.C of NPF-36. It demonstrates the progress achieved since the issuance of NPF-36 and should enable NRC Staff members to significantly reduce the number of conditions in this section of the new full power license for SNPS. Each of the seventeen (17) existing conditions is addressed.

(1) Maximum Power Level: Currently LILCO is authorized to operate SNPS at core power levels not to exceed 121.8 megawatts thermal (5%) power.

The 5% limitation is expected to be removed upon issuance of a full power license.

(2) Technical Specifications and Environmental Protection Plan

LILCO recently completed a multi-discipline review and certified that the SNPS Technical Specifications reflect the design and as-built configuration of the plant (SNRC-1559 dated March 21, 1989).

LILCO will continue to operate SNPS in accordance with the Technical Specifications and the Environmental Protection Plan.

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(3) Fire Protection Program

Generic Letter 86-10 states: "Inclusion of the fire protection program in the FSAR will be a prerequisite for Licensing for all now under review." SNPS USAR section 9.5 presently includes the SNPS fire protection program and LILCO anticipates that the standard license condition which appears on page 5 of Generic Letter 86-10 will appear in the SNPS full power license.

(4) Initial Test Program

This license condition requires LILCO to submit, within one month, any changes to the initial test program described in Section 14 of the USAR, which were made in accordance with 10 CFR 50.59. LILCO letters SNRC-1515 dated November 18, 1988 and SNRC-1520 dated December 9, 1988, were prepared in accordance with this license condition and demonstrate our compliance. From March 15 - 23, 1989, SNPS was subjected to an operational readiness assessment team inspection. On March 23, 1989, at the exit meeting, an inspector stated that a careful review of the safety evaluations which were performed to support these letters, revealed a high level of performance and a good understanding of the 10 CFR 50.59 process. LILCO will continue to only make changes to the initial test program that do not involve an unreviewed safety question and will report these in accordance with 10 CFR 50.59(b) within one month.

(5) Inservice Inspection and Testing Program

This license condition imposes a unique and unusual requirement on the NRC Staff. Paragraph a. states that the initial inservice inspection program (sent via SNRC-1135 dated May 17, 1985) will be evaluated before the first refueling outage. LILCO expects the removal of this license condition from the SNPS full power license.

Paragraph b. specifies the content and reporting requirements associated with the SNPS inservice inspection program. LILCO has complied with these requirements and believes that a detailed review and evaluation of our submittals will confirm this statement. LILCO intends to notify the NRC Staff of any significant or substantive changes in the intended inspection program, and will continue to evaluate and implement, where practical, state-of-the-art improvements in scope or methods of implementing the ISI program.

(6) Surveillance of Control Blades

This license condition requires LILCO to comply with items 1, 2 and 3 of IE Bulletin 79-26, Revision 1, entitled "Boron Loss from BWR Control Blades," and to submit a written response to item 3 within 30 days after plant startup following the first refueling outage. For the following reasons, LILCO requests the removal of this license condition from the SNPS full power license. First, LILCO revised the Boron depletion rate curve in Station Procedure 57.010.01 (Control Blade Management) to account for the boron leaching phenomenon described in the bulletin. This was done many years ago. Thus, the chemical diffusion of boron through cladding and into the reactor coolant is taken into account as LILCO determines the need for control blade replacement. Second, the reporting requirement of the bulletin was directed at operating BWRs that, up to that point in time, had not accounted for this leaching process in their depletion analysis. Thus, some utilities may have had a need to change control blades sooner than they previously thought. Third, NRC staff review of the aforementioned procedure change is complete and this bulletin was reported in Section 5 of NRC inspection report 50-322/81-20.

(7) NUREG-0737 Conditions

(a) Shift Technical Advisor

This license condition was fulfilled when NRC approved LILCO submittal SNRC-1272 dated July 11, 1986. This approval was transmitted via NRC (R. Lo) letter dated December 9, 1986 to LILCO (J. D. Leonard, Jr.). Thus, on December 9, 1986, the first group of seven (7) backup STAs were approved and this license condition was fulfilled. LILCO requests its deletion from the SNPS full power license.

(b) Control Room Design Review

This license condition requires that LILCO mark appropriate control room meters and recorders to indicate normal operating limits, trip values and alarm points. The condition originates from a 1981 LILCO submittal and since that time LILCO issued Station Procedure 21.007.03 entitled "Meter and Gauge Face Zone Markings" for use during the startup test program. The implementation of this procedure is scheduled for completion prior to the

completion of the power ascension test program. Thus, LILCO requests removal of this license condition from the SNPS full power license.

(c) Post Accident Sampling Capability

LILCO will submit to the NRC Staff modified core damage procedure(s) that include an estimation of cladding failure due to fuel overheating, as well as cladding failure due to core melt for review and approval. This procedure(s) will incorporate use of other plant parameters as indicators of core damage.

(d) Emergency Response Capabilities (Attachment 1 to NPF-36)

1. Safety Parameter Display System: LILCO currently has a fully operational interim safety parameter display system which has been found acceptable for operation during the first cycle. Prior to startup after the first refueling outage, LILCO will install, test, and train reactor operators on the use of the final safety parameter display system. Engineering of this system is essentially complete and much of the equipment is already on site.

2. Detailed Control Room Design Revisions: After meeting with NRC Staff to resolve concerns, LILCO submitted its implementation plan in December 1986. The plan was implemented as described. Only the disposition of human engineering discrepancies and appropriate implementation remains. Over four hundred (400) human engineering discrepancies were identified and are being resolved. LILCO recognizes that resolution of some of these may result in modifications. In accordance with the schedule in NPF-36, LILCO will submit the summary report with a proposed schedule for implementation prior to startup from the first refueling outage.

3. Regulatory Guide 1.97, Rev. 2: LILCO has submitted several letters describing compliance with this Regulatory Guide (SNRC-863, SNRC-1209, and SNRC-1233). NRC Staff review is complete and all concerns are resolved except for the Category 1 status of the neutron monitoring system (see safety evaluation contained in NRC (W. Butler) letter

to LILCO (J. D. Leonard, Jr.) dated September 1987). LILCO, as a member of the BWR Owners' Group, continues to request relief from the Category 1 requirements and awaits NRC response to the BWR Owners' Group topical report on this subject. Should NRC Staff disagree with the BWROG, LILCO will implement the Category 1 requirements which apply to this system in accordance with the schedule called for in the September 1987 NRC letter.

4. Implement the Upgraded Emergency Operating Procedures: LILCO revised the SNPS emergency operating procedures in accordance with Revision 4 of the BWROG Emergency Procedures Guidelines and trained all shift crews on the proper use of these new EOPs. LILCO submitted, via SNRC-1472 for NRC Staff review, a revised Procedures Generation Package, plant specific technical guidelines, an example of verification and validation, and an updated emergency operating procedure. Additionally, LILCO established a controlled copy set of SNPS emergency operating procedures for reactor operator examiners in Region I. Thus, LILCO fulfilled this license condition and requests that it be removed from the full power license.

(8) Equipment Qualification

The schedule exemption which was granted for the three (3) items listed is no longer needed. Prior to December 31, 1985, these three (3) items were environmentally qualified. Thus, LILCO requests that this license condition be removed.

(9) Instrumentation and Controls Systems Required for Safe Shutdown

LILCO will upgrade the remote shutdown panel by adding those instruments and controls identified in the license prior to startup following the first refueling. Following the modification, LILCO will conduct a procedure verification test for the remote shutdown system as-built condition. LILCO design change DOP 83-030, is currently fifty (50%) percent engineering complete. This modification will implement the following design changes:

1. Add a RHR A flow indicator;

2. Add a RBSW A flow indicator;

3. Add a SFPC pump A control;

4. Add RPV pressure (Div. 2) and suppression pool temperature (Div. 1 and 2) indicators; and

5. Upgrade quality standards for nine (9) instruments (i.e., sensors, power supplies, displays, and associated connections for RHR B flow (RPV) level and pressure, SW B header pressure, suppression pool temperature and level, RCIC flow and turbine speed and service air/N_2 pressure).

(10) Concrete and Structural Steel Internal Structures (Section 3.8.2 SER, SSER 1, SSER 3, SSER 4)

LILCO will operate the Residual Heat Removal (RHR) system in the steam condensing mode (SCM) only as a last resort under the conditions identified in 10 CFR 50.54 (x) and (y). This fact is identified in station procedures such as the RHR operating procedure SP 23.121.01 which allows the SCM only during emergency depressurization following loss of all AC power and is included in operator training programs. Thus, LILCO requests removal of this license condition from the SNPS full power license.

(11) Containment Isolation System

LILCO completed preliminary engineering work which identifies the scope and modifications needed to fulfill this license condition. However, on the basis of probabilistic risk assessment work done to date, LILCO is considering submitting a request to defer these modifications so that resources can be devoted to modifications which will substantively reduce risk to public health and safety. This written request will be submitted shortly after full power licensing. If, NRC staff denies this request, LILCO will comply with the present working of this license condition by submitting the final design output package for NRC Staff review and approval with enough time to install the additional isolation barriers prior to startup from the first refueling outage.

(12) Emergency Diesel Generator License Condition

This license condition imposes operation, maintenance, testing and inspection requirements on LILCO as a basis for satisfying the requirements of GDC 17. LILCO is committed to meeting these requirements and has acted to incorporate them into its practices and procedures.

Prior to startup following the first refueling outage, LILCO will submit the results of additional tests, inspections, and/or analysis that when considered with current nuclear industry experience on TDI diesels will provide the basis for LILCO's conclusions on whether the requirements of GDC 17 are satisfied for the second cycle and beyond.

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

In July 1983, Teledyne Engineering Services (TES) submitted a final report on their IDR of the SNPS low pressure core spray system. In this report, TES recommended a review of all studies and evaluations related to the IDR to determine where calculations should be updated to provide an appropriate set of records that can be used for maintenance, replacement, repair and modification.

This review was completed, with the results as indicated in Attachment 1, and calculations have been updated as required. Thus, LILCO requests removal of this license condition from the SNPS full power license.

(14) Seismic and Dynamic Qualification

(a) The radiation monitoring system panels and pumps identified in this license condition have been dynamically qualified. Qualification documentation was presented to NRC staff personnel during a recent inspection and LILCO has satisfactorily resolved questions associated with these qualification documents. Thus, this license condition is fulfilled and LILCO requests it be removed from the full power license.

(b) LILCO will not use the invessel rack (F16-E006/1F16*FAK-09) until dynamic qualification is complete.

(15) Operating Staff Experience Requirements

LILCO operations personnel have gained valuable operating experience since the issuance of NPF-36 on July 3, 1985. As described in the SNPS Startup Test Report and its supplements, LILCO completed several successful startups and related tests including the synchronization of the main generator to the grid for two (2) twenty-four (24) hour periods. Notwithstanding these accomplishments, LILCO will comply with the operating staff experience requirements described in Attachment 2 of NPF-36.

(16) Fission Gas Release and Ballooning Rupture

LILCO will 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 pressurized fuel, (b) accommodate the information in NUREG-0630 including its effects on local oxidation, and (c) have been reviewed and approved by the NRC. Since this type of analysis is required by 10 CFR 50.46 to support operation in the second cycle and beyond, LILCO sees no need for and requests removal of this license condition.

(17) Generic Implications of Salem ATWS Events

During a recent NRC inspection, LILCO provided the completion status of the action requirements in Generic Letter 83-28. It is repeated and updated below, as follows:

Item#	Status
1.1	Closed SSER #9
1.2	Awaiting NRC Reply to SNRC-1013 dated 3/9/84
2.1.1	Closed NRC Letter 12/29/88
2.1.2	Closed NRC Letter 12/29/88
2.2.1	Closed NRC Letter 12/28/88
2.2.2	Closed NRC Letter 12/28/88
3.1.1	Closed NRC Letter 9/19/88
3.1.2	Closed NRC Letter 9/19/88
3.1.3	Closed NRC Letter 4/9/86
3.2.1	Closed NRC Letter 9/19/88
3.2.2	Closed NRC Letter 9/19/88
3.2.3	Closed NRC Letter 4/9/86
4.5.1	LILCO Action Req. - Submit T.S. Change - Testing backup scram valves during refueling outages.
4.5.2	Closed NRC Letter 4/11/85 Attachment 4
4.5.3	Awaiting NRC reply to SNRC-1404 dated 12/29/87

In November 1988, LILCO completed a functional test of the backup scram valves and; therefore, LILCO's only

remaining action is to submit a technical specification change to include functional testing of these valves at each refueling outage. Based upon this commitment, LILCO requests removal of this license condition.

Should you or your staff require additional information, please do not hesitate to contact my office.

Very truly yours,

W. Christopher for

John D. Leonard, Jr.
Vice President - Nuclear Operations

GJG:ck

Attachments

cc: S. Brown
W. T. Russell
F. Crescenzo

J.O. No. 11600.02

Revision No. 1
October 5, 1984
SNPS-Project Procedure No. 54

Approved For J. Carroll Project Engineer 10/5/84

INCORPORATION OF INDEPENDENT DESIGN REVIEW EVALUATIONS AND STUDIES INTO APPROPRIATE CALCULATIONS SNPS-PROJECT PROCEDURE NO. 54 SHOREHAM NUCLEAR POWER STATION LONG ISLAND LIGHTING COMPANY

1.0 PURPOSE AND SCOPE

The purpose of this procedure is to ensure that whenever subsequent evaluations or formal studies are required to supplement approved calculations (related to the Teledyne Independent Design Review) that these evaluations will be appropriately incorporated into the original calculations or appropriately referenced so that when the original calculations are revised, they will benefit from data developed as a result of the Independent Design Review.

2.0 DEFINITIONS

2.1 Calculation (of record) - an engineering/design technical report that provides the basis for an engineered design or conclusion and provides the full and formal documentation of the engineering process.

2.2 Study - serves to verify or affirm the conclusions of previously established calculations. These studies are formal documentation of the calculation review process.

2.3 Review - is an informal evaluation of calculations or hardware items to determine the scope of calculations or studies required to respond to TES items.

3.0 RESPONSIBILITIES

3.1 The Lead Engineering Mechanics Engineers (pipe Stress and Pipe Supports) will be responsible for determining whether significant impact has been made to warrant revision of approved calculations or if the study or evaluation simply supports the conclusions drawn by the initial calculation (See Attachment 5.1).

4.0 PROCEDURE

4.1 Calculations that are revised as a result of supplementary evaluations and studies have been updated and the specific evaluations and studies will be referenced in the calculation, as required.

4.2 Those calculations that are not revised as a result of evaluations and studies will have this fact noted (see Attachment 5.2. The calculation index will be noted to reference the applicable study.

4.3 Those calculations or hardware items reviewed to determine the scope are listed in the attachments. Unless otherwise indicated no further actions are required for documentation of the effort.

5.0 ATTACHMENTS

5.1 Flow Chart

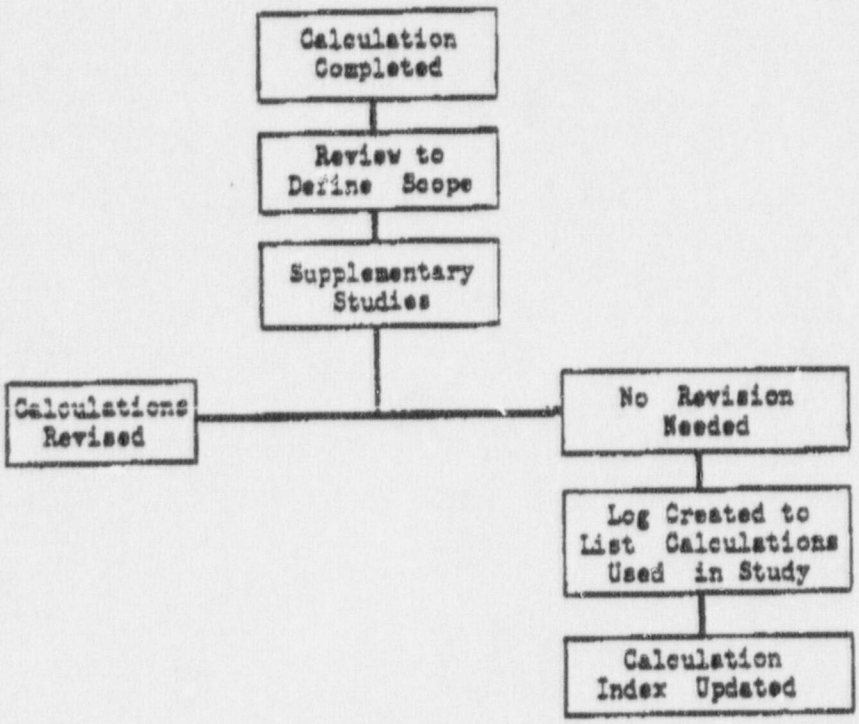
5.2 Log of Teledyne Engineering Service Review

6.0 REFERENCES

1. "Description of impact to documentation by TES ICR number," refer to Job Book No. 21 - Tes. 43, Section 1

ATTACHMENT 5.1

FLOW CHART



TELEDYNE ENGINEERING SERVICE REVIEW (CONT)

RESPONSIBLE DISCIPLINE	TES ITEM	CLASSIFICATION	SUBJECT	REVIEWS AND STUDY CALCULATIONS PERFORMED TO ADDRESS TES ITEMS	IMPACT TO DOCUMENTATION
SA	ICR # 14	Finding	Dissimiliar Metal Weld at RPV Nozzle	N/A	N/A
PWR	ICR # 15	Finding	Valve 1E21-MOV-031B Nameplate Incorrectly Marked	Reviewed Calc. 19.2.1-8-4A	Spec. SH1-88D Revised by E&DCR P-4506. Nameplate Modified by LILCO in accordance with LIL-23552
SA	ICR # 16	Observation	Temperature Used for Calculating RPV Nozzle Displacements	11600.02-NP(B)-615-X5	No Impact on Existing Design Calculations, Calculation Index for AX 25J, 1B annotated
PS	ICR # 17	Finding	Interface between Revisions to Pipe Suppt Drawings and Calculatins	N/A	N/A
PSAS	ICR # 18	Finding with Additional Concerns	Combination of Time History loads with Other Loads for Pipe Support Design	Review	Calculations revised
SA	ICR # 19	Finding with Additional Concerns	Core Spray Test Mode Time History Analysis	Review	Calculations revised
SA	ICR # 20	Finding	Tie-Back Support PSR-5205 (AX-10D) Not Shown on ISO # IC-66	N/A	N/A
MECH	ICR # 21	Finding with Additional Concerns	Valve Accelerations at Motor Operated Valves	11600.02-NM(B)-399-CZC 11600.02-NM(B)-683-XD 11600.02-NP(B)-716-XD 11600.02-NP(B)-725-XD	Calculation Index for NM(B)-254, AX 11A, 524AZ annotated
PS	ICR # 22	Observation	Corner and Lada Sprin Used for Bergen-Paterson Support	N/A	N/A
—	ICR # 23	CLOSED	Item Closed	N/A	N/A
SA	ICR # 24	Observation	NUPIPE Use of Load Set Pairs	N/A	N/A

TELEDYNE ENGINEERING SERVICE REVIEW (CONT)

SNPS-PP54 Rev. 1 Attachment 5.2

RESPONSIBLE DISCIPLINE	TES ITEM	CLASSIFICATION	SUBJECT	REVIEWS AND STUDY CALCULATIONS PERFORMED TO ADDRESS TES ITEMS	IMPACT TO DOCUMENTATION
PS	ICR # 25	Observation	Calc. IE2i-psr062 Log Calculation is Incorrect	Review	Calculation revised
STRUCT	ICR # 26	Observation	Qualification of Penetration X-20	N/A	Calculation revised
SA	ICR # 27	Finding with Additional Concerns	S.I.F. at Small Bore Connection to large Piping	11600.02-NP(B)-647-XH (CL2/3 on CL1); 11600.02-NP(B)-644-XH (CL2/3)	Adversely affected CL1 calculations revised. CL2/3 calculations not affected
SA	ICR # 28	Finding with Additional Concerns	Temperature Distribution Modeling of Tie-Back Supports	11600.02-NP(B)-673-XC	Adversely affected calculations revised