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December 4, 1984 SNRC-1116

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

Generic Letter 83-28, "Required Actions
Based on Generic Implications of Salem ATWS Events"
Submittal of Additional Information
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Reference 1: LILCO letter, SNRC-1013 (J. L. Smith) to the NRC (H. R. Denton) dated March 9, 1984

Dear Mr. Denton:

The initial LILCO response to Generic Letter 83-28 (Reference 1) deferred the submittal of positions on several items pending the completion of the BWR Owners' Group and INPO efforts, and the SNPS specific evaluation thereof. The LILCO responses to items 1.1.1, 1.1.3, 1.1.5, 2.1.1, 2.2.2 and 4.5 are enclosed herewith as Attachment 1.

Very truly yours,

Jeffrey A. Smith P.O.V

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

RJT:ck

Attachments

cc: P. Eselgroth

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ATTACHMENT 1

Item 1.1.1 CRITERIA FOR THE JUSTIFICATION OF RESTART

The occurrence of an unanticipated scram at a nuclear power plant should not in and of itself be the basis for an extended reevaluation of the operation of the engineered safety features related to safe operation. The purpose of this section is to show that there is a sound rationale, based on technical judgement, for the decision on the part of senior experienced operating plant personnel to exercise and justify a restart of the plant, predicated upon: (1) operator knowledge of the plant; (2) existing plant procedures, and (3) control room indicators. This technical judgement shall ensure that the following restart criteria are met:

a. The plant is shown to be in a safe condition.

b. The cause of the event is either understood, or, after a detailed investigation, is considered to have been a spurious trip with a reasonably low potential for reoccurrence.

c. The need for corrective action has been determined and appropriately implemented.

d. The expected on-off automatic operation of plant safety related systems has been observed.

e. The approval of responsible Utility management has been obtained.

Criterion a

The determination of the safe condition of the plant is assumed for the purpose of this discussion. Therefore, for the remainder of this discussion, it will be assumed that the safety limits have not been violated or exceeded and the issue at hand is one of justifying restart from a normal shutdown condition. If safety limits have not been violated and the plant is in a stable and safe mode, then the operator may begin evaluation of the advisability of restart.

Criterion b

The operator has many sources of information available to him which can be used both as a diagnostic tool in evaluating the cause of an unanticipated scram and in the identification of other than expected performance of plant systems and equipment. The readout of both safety related and non-safety related indicators (including, as applicable, such sources as the process computer, alarm typer, trend recorder, and sequence of event recorder) provide a basis upon which a technically defensible action can be

initiated to determine the cause of the event, and to assure that the cause of the scram no longer exists. See Caution No. 3 of the BWR Emergency Procedure Guidelines (EPGs) and LILCO's previous submittal, SNRC-1013 (3/9/84), Items 1.1.4 and 1.2.

It is important to understand the cause of an unscheduled trip so that a reoccurrence can be avoided after restart. However, it is not realistic to ignore the possibility of spurious trips whose cause cannot be positively identified.

If the cause of the unscheduled reactor shutdown cannot be determined, and the safety systems have indicated a proper response, the Review of Operations Committee (ROC) may recommend a restart based on the following conditions:

a. No physical damage was done by the event and a determination was made that the plant had not operated beyond the boundaries established by approved plant safety and transient analysis.

b. Reasonable action to gain additional information has been considered.

The ultimate responsibility for the restart decision rests with the Plant Manager.

The discussion of the qualifications and responsibilities of the personnel making the restart recommendation is presented in SNRC-1013, Items 1.1.2, and 1.1.6, Chapter 13 of the SNPS FSAR and Section 1.1.3 of this letter.

Criterion c

Once the cause of the event is determined, using Control Room supplied information, the operator then needs to determine what, if any, corrective action(s) needs to be implemented. The Institute of Nuclear Power Operation Good Practices (OP-211) contains the following criteria which are relevant to this course of action:

Condition I
The cause of the trip is positively known and has been corrected; all safety-related and other important equipment functioned properly during the trip, or if not, the malfunction has either been corrected or redundant equipment is available for reactor start-up.

Condition II
The cause of the trip is not positively known or safety-related or other important equipment functioned in an abnormal or degraded manner during the trip and the malfunction has not been corrected or redundant equipment is not available to allow start-up.

Demonstrating compliance to these criteria justifies the initiation of restart activities. When the cause of the scram is determined, a decision can be made on the need for corrective action. Such a decision can fall into three categories: (a) no corrective action is required; (b) corrective action is required but does not need to be performed before restart (i.e., Technical Specifications do not require actions as a condition of restart); or (c) corrective action is required before restart.

If no action is required (i.e., readouts in the control room are within allowable limits), normal restart procedures apply. If a corrective action is required, but that action is not required in order to meet Technical Specification limits, then restart procedures apply and the needed corrective actions are initiated following restart. If corrective action is required, then it would be necessary to complete the effort before initiation of restart activities. This action can range from a simple recalibration of the device executing the scram to replacement and/or recalibration of a system. This determination also needs to be based on the Technical Specification associated with startup activities (i.e., Technical Specifications allow restart with some devices out-of-service). Before startup activities are commenced, compliance to the Technical Specifications must be assured. Also, assurance must be provided that, as a result of the investigation in the event, matters such as valve alignments are brought back into the proper sequence and/or arrangements.

Criterion d

If the operator determines that a particular system should have been initiated for a particular event, we need only establish that the system did indeed initiate and in the proper sequence. A detailed analysis of the actual performance of that system following an unscheduled shutdown is not a criterion for restart. Such a detailed analysis is accomplished through the normal surveillance testing procedure done at regular intervals. This step is consistent with the philosophy espoused in Caution No. 1 of the NRC approved BWR EPGs. Confidence in the accuracy of control room readout is provided both by the routine maintenance and surveillance activities associated with engineered safety features, and normally scheduled and performed calibration activities associated with such devices. Adherence to these efforts mitigates against the need to enter into a complete recalibration (i.e., pressure, flow, operating times, etc.) or performance reevaluation of the adequacy of system operation.

Criterion e

The recommendation to restart will not be left solely to the determination of the senior operator. Based upon the applicable plant specific and administrative procedures controlling plant operation, the review and analysis of unscheduled reactor shutdowns will be performed by the Watch Engineer with input from the Shift Technical Advisor. The Operating Engineer will ensure that

the unscheduled shutdown has been analyzed and the cause determined and corrected prior to authorizing a restart. If the cause of the event cannot be positively identified, the Operating Engineer will consult the Review of Operations Committee (SNRC-1013, Item 1.1.2). The recommendation for restart requires the concurrence of the Operations Manager. The Plant Manager provides the authorization for restart after any unscheduled reactor shutdown.

Item 1.1.3 PERSONNEL QUALIFICATIONS

The analyses of unscheduled shutdowns at BWR power plants are performed by a select group of trained and qualified individuals. The education, training, and job related experience of these people provide an immense store of knowledge forming the basis for the evaluation needed to make the post-trip review and restart recommendation.

Utilizing the Shoreham specific procedure that has been established for dealing with unscheduled shutdown conditions, steps will be taken toward making a decision to restart. At SNPS the responsible personnel are, as a minimum, the Watch Engineer, the Shift Technical Advisor (STA), Operating Engineer and the Operations Manager. The restart recommendation is then made to the Plant Manager for the final decision to restart.

The qualifications and training requirements of these personnel are described in Chapter 13 of the SNPS FSAR. The Operating Engineer, has as a minimum, the experience qualifications listed for Operations Manager in ANSI N18.1-1971 and holds a Senior Reactor Operators (SRO) License. The Watch Engineer has a high school diploma or equivalent, 4 years of power plant experience and also holds a SRO License. The Shift Technical Advisor has a bachelor's degree in a scientific or an engineering discipline and has received specialized training in thermohydraulics, heat transfer and accident/transient analysis.

The aforementioned holders of the NRC SRO licenses have qualifications that have to be met concerning examination of plant systems and operations, Technical Specifications, and transients. The STA is also qualified to the recommendations of NUREG-0737, which requires training in the analysis and determination of causes of off-normal situations.

Along with all the initial training requirements for these positions, retraining and periodic requalification, as required by regulation, are performed. Retraining examines plant operations and off-normal events while requalification ensures that the recommended degree of knowledge is being maintained.

In addressing unscheduled shutdowns and the inherent complexities involved, the SNPS procedure also includes those cases where, should uncertainty exist regarding a safe restart, additional expertise is called upon to reach a technically defensible

position. Provisions are made for obtaining information from additional qualified plant personnel to provide more specific and detailed information if needed. In addition to the Review of Operations Committee, these people can include other licensed personnel, plant support personnel knowledgeable in specific areas and varying levels of plant management. Specific input is available from cognizant persons regarding areas of maintenance and instrumentation. The information provided by these people is based on their experience, organizational responsibilities, and training that they have received in those specific areas. This combined expertise provides the necessary technically based information to enable a recommendation to be made concerning restart.

Item 1.1.5 EVENT INFORMATION COMPARISON CRITERIA

Historical data from the process computer and chart recorders will be compared with the trip settings for reactor protection, emergency core cooling system and group isolations contained in the Technical Specifications to ensure equipment has operated as designed. When applicable, the transient will be compared with the transients analyzed in the FSAR to confirm that plant behavior was consistent with calculated values.

Item 2.1.1 EQUIPMENT CLASSIFICATION (Reactor Trip System Components)

LILCO has reviewed the BWR Owner's Group Reactor Trip System (RTS) Equipment Classification Methodology as presented in NEDC 30505, "Response Guidelines for NRC Generic Letter 83-28", dated March, 1984. The Shoreham-specific RTS Component List (SNRC-1013, Item 2.1.1) is in general agreement with the Owner's Group position. Based on this review several additional components are in the process of being added to the aforementioned list.

Item 2.1.1 VENDOR INTERFACE (Reactor Trip System)

In response to this concern, LILCO joined with 55 other utilities and formed an Institute of Nuclear Power Operations (INPO) Nuclear Utility Task Action Committee (NUTAC). This committee has developed and approved an industry-wide Vendor Equipment Technical Information Program (VETIP), which is described in detail in Attachment 2. This program promotes interaction among the major organizations involved in the generation of commercial nuclear power. As illustrated in Figure 1 to the previously mentioned attachment, individual Utilities exchange and disseminate safety related system and component information with vendors, the NRC, INPO and other Utilities. This exchange of information takes place via written notification (i.e., Licensee Event Reports, NRC I&E Bulletins and Information Notices, industry newsletters, etc.) as well as industry meetings and day-to-day verbal communications. The purpose of these information exchanges is to share equipment technical information to improve the safety and reliability of nuclear power generating stations. The primary purpose of the VETIP program is to ensure that current information and data will

be made available to those personnel responsible for developing and maintaining plant instructions and procedures. These information systems and programs currently exist and are capable of identifying to the industry precursors that could lead to a Salem-type event. It should be noted that the VETIP is industry-controlled and is mainly a hardware oriented program that does not rely on vendor action, other than the NSSS supplier, to provide information directly to Utilities. Instead, the VETIP provides information developed by industry experience through Significant Event Reports (SERs) and Significant Operating Experience Reports (SOERs) to the equipment vendor for comment before it is circulated to the Utilities concerned.

LILCO is a recipient of information from existing General Electric Company (our NSSS vendor) service advisory programs. This program consists of two major categories: (a) information regarding safety-related systems and components; and (b) technical information intended to enhance safety and non-safety related equipment reliability and improve plant performance. These programs include, but are not limited to:

(a) 10CFR21 Reporting. The General Electric Company has established a reporting system to handle safety concerns that complies with the requirements of 10CFR21.

(b) Customer (Urgent Communications). In addition to the 10CFR21 reports, a procedure for handling urgent communications to BWR owner/operators has been established for use in providing fast notification of safety concerns. These communications are usually in the form of a short letter which provides a brief explanation and advice or precautionary measures to be observed to avoid potential operational hazards. As a result of their urgent nature, these communications are processed to operating plants by the most effective method, either by telephone or, if transmitted in written form, they will be followed up or preceded by a telephone call.

The programs for SILs and TILS include a mechanism to ensure receipt of the GE information. As described in our previous response (SNRC-1013, Item 2.1.1) plant procedures are in place to receive, control, review and utilize this information.

Item 2.2.2 VENDOR INTERFACE

LILCO formally endorses the Nuclear Utility Task Action Committee (NUTAC) Vendor Equipment Technical Information Program (VETIP) which is enclosed as Attachment 2 for your information.

LILCO considers VETIP to effectively address the concerns raised by Item 2.2.2 of the Generic Letter.

The administrative program and procedures now in effect at Shoreham have been reviewed to confirm that they implement the intent of the recommended guidelines presented in the NUTAC report.

Items 4.5.1 and 4.5.2 REACTOR TRIP SYSTEM FUNCTIONAL TESTING

System Design

The Shoreham Reactor Protection System (RPS) is designed to provide the highest practical degree of plant safety, with continuity of service as the second basic criterion. The RPS design is based on two (2) separate trip systems, trip systems A and B. Each of the trip systems has two (2) independent trip channels, sensors, and associated equipment for each measured variable. The RPS provides an optimum safe compromise between reliability and freedom from spurious scrams. There is a negligible probability of failure to scram on a predetermined scram condition and a very low probability of a spurious scram under normal operating condition. The RPS is of the dual-trip, fail-safe type which requires a trip of both trip systems to initiate a plant shutdown.

There are 137 HCUs (Hydraulic Control Units) at Shoreham, one for each control rod drive. Each control rod drive is individually operated so that one failure does not affect the operation of any other control rod drive. Each control rod is provided with two 3-way solenoid operated scram pilot valves and two scram valves. The solenoids of the pilot valves are energized by outputs from the two trip channels A & B of the RPS.

Two redundant back up scram valves are provided in G.E. plants to assure that the control rods actuate should any of the pilot scram valves fail to function. These backup valves are also interconnected with the RPS to ensure that discharge volume vent and drain valves close (and that the HCU scram valves open) upon a scram signal in the event of failure of the pilot valves (or failure of the scram pilot air valves in one or more HCUs) by dumping the air header supply pressure.

Functional Testing

The following is a summary of the on-line functional testing and testing intervals performed on the RPS at Shoreham:

Consistent with the Technical Specifications, on-line channel functional testing is performed on the multiple and diverse reactor transient trip sensors [Average Power Range Monitor (APRM) and Intermediate Range Monitor (IRM) Reactor trip signal channels, and multiple and diverse Scram Discharge Volume High water level trips]. During the required trip sensor channel tests discussed above, each scram contactor which actuates the scram pilot solenoid valves is tested. The simple operation of the scram contactors minimizes concerns of wear, and frequent testing assures that any failures are detected early. The Scram Pilot Solenoid Valves which are actuated by the scram contactors are all tested regularly.

Redundant Electrical Protection Assemblies (EPAs) which protect the Scram Pilot Solenoid Valves from low voltage chattering (and the associated potential consequence of accelerated wear) are also functionally tested.

The surveillance testing requirements related to the Scram Pilot Solenoid Valves assure that the probability of undetected failure of these independently acting solenoid valves is small.

Channel functional tests are performed on-line for the following sensor trips:

- Reactor Vessel Dome Pressure-High
- Reactor Vessel Water Level-Low
- Main Steam Line Isolation Valve-Closure
- Main Steam Line Radiation-High
- Drywell Pressure-High
- Turbine Control Valve Fast Closure, Control Oil Pressure-Low
- Turbine Stop Valve-Closure

Channel functional tests are also performed for APRMS and IRMS.

Each sensor channel functional test includes full actuation of the associated logic, the two output scram contactors in each channel, and the individual CRD scram air pilot valve solenoids for the associated logic division (solenoids from both logic Division A and B are required for scram initiation).

Most credible failures within the RPS logic will de-energize a set of scram solenoids which causes a half scram, i.e., one of the two scram solenoids required for scram initiation is de-energized at some or all hydraulic control units. These failures would be "SAFE" failures that would increase the probability of plant shutdown.

The less credible logic failures which prevent a channel from de-energizing will be detected during channel functional testing in compliance with Technical Specification requirements. The tests described above ensure that an increase in failure rate due to a wearout condition or a common cause failure potential could be detected early and corrective action taken before the failure condition becomes systematic.

Other channel functional tests include testing on the Scram Discharge Volume (SDV) Water Level-High trip and manual scram trip and test of the reactor mode switch in the shutdown position every refueling. The first two trips involve on-line testing and the latter mode switch test can only be conducted during reactor shutdown. The manual scram trip can be tested on-line without creating a scram.

The testing of the SDV Water Level-High trip is considered adequate based on the current designed redundancy and diversity incorporated into the system.

There are two diverse and redundant sets of level sensors which scram the reactor in the unlikely event of high water level in the SDV during power operation. These trips are designed to allow sufficient scram water discharge volume given the scram trip point is reached.

Remote manual switches in the pilot valve solenoid circuits allow the discharge volume vent and drain valves to be tested without disturbing the reactor protection system. At Shoreham, redundant vent and drain valves are provided to ensure against loss of reactor coolant from the SDV following a scram. Closing the SDV valves allows the outlet scram valve seats to be leak-tested by timing the accumulation of leakage inside the scram discharge volume.

The "BWR Scram System Reliability Analysis" (Reference 1) concluded that reactor shutdown can be achieved if at least 50% of the control rods in a checkerboard pattern and 69% in a random pattern are inserted in the core. The probability of independent failure of enough rods to prevent shutdown is very small. The most unlikely type of failure would be some common cause mechanism that if undetected over a long period of time would cause unsafe shutdown. The Technical Specification surveillance requirements adequately ensure that a failure mechanism affecting several individual drives (considered to be very remote) would not go undetected.

One of the major features that ensures that several drives do not fail at one time due to wearout or a common cause is the staggered maintenance and overhaul of selected CRDs or Hydraulic Control Units (HCUs) at refueling outages. This ensures a mix of drives by age, component lot, maintenance time and servicing personnel, and testing.

The scram insertion time tests include, in addition to drive timing and insertion capability, a test of operability of the HCU scram insert and discharge valves including associated scram air pilot valves. As stated in the previous paragraph, the required frequency of testing given in the Technical Specification ensures that a systematic failure mechanism in the HCUs would be detected early enough and corrective action taken before the condition becomes a critical failure preventing scram.

In order to address the ATWS issue prior to fuel load LILCO formally committed to the NRC (SNRC-437, dated 10/19/79) to perform several modifications which will meet the requirements of NUREG-0460, Vol. 4, alternative 2A. A portion of this task involved the implementation of ARI to provide Alternate Rod Insertion in the unlikely event of a failure of the reactor protection system. At Shoreham ARI has been implemented by the addition of two (2) Cat I (safety related) solenoid operated pilot air valves. The solenoids of these valves are energized from (ATWS) logic instrumentation diverse from that of the backup scram valves which are energized from the RSP logic.

The following apply to Shoreham's ARI:

- INITIATES ON LOW REACTOR WATER LEVEL OR HIGH REACTOR PRESSURE
- ELECTRICALLY INDEPENDENT FROM RPS
- LOCATED ON SCRAM PNEUMATIC HEADER
- INDEPENDENT LOGIC FROM RPS, ALSO UTILIZED BY RPT
- UTILIZES "MONTICELLO" LOGIC
- LOGIC IS "ENERGIZE TO OPERATE"

In order to further insure system reliability the non-safety related backup scram solenoid valves initially provided by G.E. were replaced by qualified safety related solenoid valves.

In summary, the current RPS on-line surveillance requirement; the multiple and diverse scram sensors; the modification of the SNPS CRD System to provide an Alternate Rod Insertion capability and the use of qualified safety related backup scram valves ensures that the probability of the failure of enough control rods to prevent a scram is negligible. Functional testing of these valves during plant operation would increase the likelihood of a plant scram resulting in a significant challenge to plant safety systems and a degradation of plant safety. LILCO therefore requests exemption from the requirement for on-line functional testing of the SNPS backup scram valves.

Reference

1. "BWR Scram System Reliability Analysis", W. P. Sullivan, et al, September 30, 1976 (Transmitted in letter from E. A. Hughes (GE) to D. F. Ross (NRC), "General Electric Company ATWS Reliability Report," September 30, 1976).

Item 4.5.3 Review of Technical Specification Testing Intervals

LILCO is a non-voting member of the BWROG Technical Specification Improvement Committee (TSIC). Long Island Lighting Company intends to utilize this membership to remain cognizant of the Owners' Group positions related to the existing intervals for Technical Specification mandated, on-line functional testing. A formal LILCO position regarding any BWROG TSIC recommendations will be submitted to the NRC following an internal, Shoreham specific, evaluation.