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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 9 1980

Docket No.: 50-322

Mr. Andrew W. Wofford
Vice President
Long Island Lighting Company
175 East Old Country Road
Hicksville, New York 11801

Dear Mr. Wofford:

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION - SHOREHAM NUCLEAR POWER STATION

As a result of our review of your application for an operating license for the Shoreham Nuclear Power Station, we find that we need additional information in the areas of Mechanical Engineering, Power Systems, and Instrumentation and Control. The specific information required is provided in the Enclosure.

If you desire any discussion or clarification of the information requested, please contact J. N. Wilson, Licensing Project Manager, (301) 492-8408.

Sincerely,

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
Request for Additional
Information

ccs w/enclosures:
See next page

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REQUEST FOR ADDITIONAL INFORMATION

112.23 10 CFR 50.55a has recently been revised with respect to pump and valve inservice testing requirements (See October 9, 1979 Federal Register, pp. 57912-4)

Provide a program for initial 120 month inservice testing of pumps and valves, as required by 10 CFR 50.55a(g)(4)(i). The applicable code for this inspection interval which would be required by 10 CFR 50.55a(g)(4)(i) is the Code endorsed by 10 CFR 50.55a(b)(2) 12 months prior to the date of issuance of your OL. Effective November 1, 1979, 10 CFR 50.55a(b)(2) endorsed the 1977 Edition with all agenda through Summer 1978. Your IST program should be prepared in accordance with the attached guidance and should indicate which code requirements are impractical to meet together with documentation for justification why relief is necessary.

NRC STAFF COMMENTS ON INSERVICE PUMP AND VALVE TESTING PROGRAMS AND RELIEF REQUESTS

The NRC staff, after reviewing a number of pump and valve testing programs, has determined that further guidance might be helpful to illustrate the type and extent of information we feel is necessary to expedite the review of these programs. We feel that the Licensee can, by incorporating these guidelines into each program submittal, reduce considerably the staff's review time and time spent by the Licensee in responding to NRC staff requests for additional information.

The pump testing program should include all safety related* Class 1, 2, and 3 pumps which are installed in water cooled nuclear power plants and which are provided with an emergency power source.

The valve testing program should include all the safety related valves in the following systems excluding valves used for operating convenience only, such as manual vent, drain, instrument, and test valves, and valves used for maintenance only.

PWR

a. High Pressure Injection System

b. Low Pressure Injection System

c. Accumulator Systems

d. Containment Spray System

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*Safety related - necessary to safely shut down the plant and mitigate the consequences of an accident.

e. Primary and Secondary System Safety and Relief Valves

f. Auxiliary Feedwater Systems

g. Reactor Building Cooling System

h. Active Components in Service Water and Instrument Air Systems which are required to support safety system functions.

i. Containment Isolation Valves required to change position to isolate containment.

j. Chemical & Volume Control System

k. Other key components in Auxiliary Systems which are required to directly support plant shutdown or safety system function.

l. Residual Heat Removal System

m. Reactor Coolant System

BWR

a. High Pressure Core Injection System

b. Low Pressure Core Injection System

c. Residual Heat Removal System (Shutdown Cooling System)

d. Emergency Condenser System (Isolation Condenser System)

e. Low Pressure Core Spray System

f. Containment Spray System

g. Safety, Relief, and Safety/Relief Valves

h. RCIC (Reactor Core Isolation Cooling) System

i. Containment Cooling System

j. Containment isolation valves required to change position to isolate containment.

k. Standby liquid control system (Boron System)

l. Automatic Depressurization System (any pilot or control valves, associated hydraulic or pneumatic systems, etc.)

m. Control Rod Drive Hydraulic System ("Scram" function)

n. other key components in Auxiliary Systems which are required to directly support plant shutdown or safety system function.

o. Reactor Coolant System

Inservice Pump and Valve Testing Program

I. Information required for NRC Staff Review of the Pump and Valve Testing Program

A. Three sets of P&ID's, which include all of the systems listed above, with the code class and system boundaries clearly marked. The drawings should include all of the components present at the time of submittal and a legend of the P&ID symbols.

B. Identification of the applicable ASME Code Edition and Addenda

C. The period for which the program is applicable.

D. Identify the component code class.

E. For Pump testing: Identify

1. Each pump required to be tested (name and number)
2. The test parameters to be measured
3. The test frequency

F. For valve testing: Identify

1. Each valve in ASME Section XI Categories A & B that will be exercised every three months during normal plant operation (indicate whether partial or full stroke exercise, and for power operated valves list the limiting value for stroke time.)

2. Each valve in ASME Section XI Category A that will be leak tested during refueling outages (Indicate the leak test procedure you intend to use)

3. Each valve in ASME Section XI Categories C, D, and E that will be tested, the type of test and the test frequency. For check valves, identify those that will be exercised every 3 months and those that will only be exercised during cold shutdown or refueling outages.

II. Additional Information that will be Helpful in Speeding Up the Review Process

A. Include the valve location coordinates or other appropriate location information which will expedite our locating the valves on the P&IDs.

B. Provide P&ID drawings that are large and clear enough to be read easily.

C. Identify valves tht are provided with an interlock to other components and a brief description of that function.

Relief Requests from Section XI Requirements

The largest area of concern for the NRC staff, in the review of an inservice valve and pump testing program, is in evaluating the basis for justifying relief from Section XI Requirements. It has been our experience that many requests for relief, submitted in these programs, do not provide adequate descriptive and detailed technical information. This explicit information is necessary to provide reasonable assurance that the burden imposed on the licensee in complying with the code requirements is not justified by the increased level of safety obtained.

Relief requests which are submitted with a justification such as "Impractical", "Inaccessible", or any other categorical basis, will require additional information, as illustrated in the enclosed examples, to allow our staff to make an evaluation of that relief request. The intention of this guidance is to illustrate the content and extent of information required by the NRC staff, in the request for relief, to make a proper evaluation and adequately document the basis for that relief in our safety evaluation report. The NRC staff feels that by receiving this information in the program submittal, subsequent requests for additional information and delays in completing our review can be considerably reduced or eliminated.

I. Information Required for NRC Review of Relief Requests

A. Identify component for which relief is requested:

1. Name and number as given in FSAR
2. Function
3. ASME Section III Code Class
4. For valve testing, also specify the ASME Section XI valve category as defined in IWV-2000

B. Specifically identify the ASME Code requirement that has been determined to be impractical for each component.

C. Provide information to support the determination that the requirement in (B) is impractical; i.e., state and explain the basis for requesting relief.

D. Specify the inservice testing that will be performed in lieu of the ASME Code Section XI requirements.

E. Provide the schedule for implementation of the procedure(s) in (D).

II. Examples to Illustrate Several Possible Areas Where Relief May Be Granted and the Extent and Content of Information Necessary to Make An Evaluation

A. Accessibility: The regulation specifically grants relief from the code requirement because of insufficient access provisions. However, a detailed discussion of actual physical arrangement of the component in question to illustrate the insufficiency of space for conducting the required test is necessary.

Discuss in detail the physical arrangement of the component in question to demonstrate that there is not sufficient space to perform the code required inservice testing.

What alternative surveillance means which will provide an acceptable level of safety have you considered and why are these means not feasible?

B. Environmental Conditions (e.g., High radiation level, High temperature, High humidity, etc.)

Although it is prudent to maintain occupation radiation exposure for inspection personnel as low as practicable, the request for relief from the code requirements cannot be granted solely on the basis of high radiation levels alone. A balanced judgment between the hardships and compensating increase in the level of safety should be carefully established. If the health and safety of the public dictates the necessity of inservice testing, alternative means or even decontamination of the plant if necessary should be provided or developed.

Provide additional information regarding the radiation levels at the required test location. What alternative testing techniques which will provide an acceptable level of assurance of the integrity of the component in question have you considered and why are these techniques determined to be impractical?

C. Instrumentation is not originally provided

Provide information to justify that compliance with the code requirements would result in undue burden or hardships without a compensating increase in the level of plant safety. What alternative testing methods which will provide an acceptable level of safety have you considered and why are these methods determined to be impractical?

D. Valve Cycling During Plant Operation Could Put the Plant in an Unsafe Condition

The licensee should explain in detail why exercising tests during plant operation could jeopardize the plant safety.

E. Valve Testing at Cold Shutdown or Refueling Intervals in Lieu of the 3 Month Required Interval

The licensee should explain in detail why each valve cannot be exercised during normal operation. Also, for the valves where a refueling interval is indicated, explain in detail why each valve cannot be exercised during cold shutdown intervals.

III. Acceptance Criteria for Relief Request

The Licensee must sucessfully demonstrate that:

1. Compliance with the code requirements would result in hardships or unusual difficulties without a compensating increase in the level of safety and noncompliance will provide an acceptable level of quality and safety, or

2. Proposed alternatives to the code requirements or portions thereof will provide an acceptable level of quality and safety.

Standard Format

A standard format, for the valve portion of the pump and valve testing program and relief requests, is included as an attachment to this Guidance. The NRC staff believes that this standard format will reduce the time spent by both the staff in our review and by the licensee in their preparation of the pump and valve testing program and submittals. The standard format includes examples of relief requests which are intended to illustrate the application of the standard format and are not necessarily a specific plant relief request.

ATTACHMENT

STANDARD FORMAT

VALVE INSERVICE TESTING PROGRAM SUBMITTAL

Valve Number	Class	Coordinates	Valve Category A	B	C	D	E	Size (inches)	Valve Type	Actuator Type	Normal Position	Test Requirements	Relief Requests*	Testing Alternative	REMARKS (Not to be used for relief basis)
710	3	D-14					X	4	GA	M	LO	ET			
700	3	D-15				X		6	DE	NA	C	DT			
717	3	C-15			X			16	CK	SA	-	CV	X	CS	
702C	3	C-15			X			16	CK	SA	-	CV			
707	3	E-14			X			3	REL	SA	-	CV			
834	3	D-11		X			X	4	GL	M	C	Q	X	ET	
												MT			60 sec.
722B	3	B-11			X			3/4	REL	SA	-	SRV			
722C	3	B-11			X			3/4	REL	SA	-	SRV			
715	2	A-10			X			3	REL	SA	-	SRV			
729	2	B-10			X			3	REL	SA	-	SRV			
744B	2	D-14	X					10	GA	MO	C	Q			
												LT	X		
												MT			30 sec.

Legend for Valve Testing Exercise Frequ[illegible]

Q - Exercise valve (full stroke) for operability every (3) months

LT - Valves are leak tested per Section XI Article IWV-3420

MT - Stroke time measurements are taken and compared to the stroke time limiting value per Section XI Article IWV 3410

CV - Exercise check valves to the position required to fulfill their function every (3) months

SRV - Safety and relief valves are tested per Section XI Article IWV-3510

DT - Test category D valves per Section XI Article IWV-3600

ET - Verify and record valve position before operations are performed and after operations are completed, and verify that valve is locked or sealed.

CS - Exercise valve for operability every cold shutdown

RR - Exercise valve for operability every reactor refueling

System: Auxiliary Coolant System, Component Cooling

1. Valve: 717

Category: C

Class: 3

Function: Prevent backflow from the reactor coolant pump cooling coils

Impractical test requirement: Exercise valve for operability every three months

Basis for relief: To test this valve would require interruption of cooling water to the reactor coolant pumps motor cooling coils. This action could result in damage to the reactor coolant pumps and thus place the plant in an unsafe mode of operation.

Alternative Testing: This valve will be exercised for operability during cold shutdowns.

2. Valve: 834

Category: B-E

Class: 3

Function: Isolate the primary water from the component cooling surge tank during plant opertion. It is normally in the closed position, but routine operation of this valve will occur during refueling and cold shutdowns.

Impractical Test Requirement: Exercise valve (full stroke) for operability every three (3) months.

Basis for Relief: This valve is not required to change position during plant operation to accomplish its safety function. Exercising this valve will increase the possibility of surge tank line contamination.

Alternate Testing: Verify and record valve position before and after each valve operation.

3. Valve: 744B

Category: A

Class: 2

Function: Isolate the residual heat exchangers from the cold leg R.C.S. backf ow and accumulator backflow.

Test Requirements: Seat leakage test

Basis for Relief: This valve is located in a high radiation field (2000 mr/hr) which would make the required seat leakage test hazardous to test personnel. We intend to seat leak test two other valves (875B and 876B) which are in series with this valve and will also prevent backflow. We feel that by complying with the seat leakage requirements we will not achieve a compensatory increase in the level of safety.

Alternative Testing: No alternative seat leak testing is proposed.

223.88 Several instances have been reported where automatic closure of the containment ventilation/purge valves would not have occurred because the safety actuation signals were either manually overriden or bypassed (blocked) during normal plant operations. In addition, a related design deficiency with regard to the resetting of engineered safety feature actuation signals has been found at several operating facilities where, upon the reset of an ESF signal, certain safety related equipment would return to its non-safety mode.

Specifically, on June 25, 1978, Northeast Nuclear Energy Company discovered that intermittent containment purge operations had been conducted at Millstone Unit No. 2 with the safety actuation signals to redundant containment purge isolation valves (48 inch butterfly valves) manually overriden and inoperable. The isolation signals which are required to automatically close the purge valves to assure containment integrity were manually overriden to allow purging of containment with a high radiation signal present. The manual override circuitry designed by the plant's architect/engineer defeated not only the high radiation signal but also all other isolation signals to these valves. To manually override a safety actuation signal, the operator cycles the valve control switch to the closed position and then to the open position. This action energized a relay which

blocked the safety signal and all [illegible] safety actuation signal. This circuitry [illegible] certain valves after an accident to allow manual operation of required safety equipment.

On September 8, 1978, the staff was advised that, as a matter of routine, Salem Unit No. 1 had been venting the containment through the containment ventilation system valves to reduce pressure. In certain instances this venting has occurred with the containment high particulate radiation monitor isolation signal to the purge and pressure-vacuum relief valves overridden. The override of this containment isolation signal was accomplished by resetting the train A and B reset buttons. Under these circumstances, six valves in the containment vent and purge systems could be opened with the radiation isolation signal present. This override was performed after verifying that the actual containment particulate levels were acceptable for venting. The licensee, after further investigation of this practice, determined that the reset of the particulate radiation monitor alarm also overrides the containment isolation signal to the purge valves such that the purge valves would not have automatically closed on an emergency core cooling system (ECCS) safety injection signal.

A related design deficiency was discovered during a review of system operation following a recent unit trip and subsequent safety injection at North Anna No. 1. Specifically, it was found that certain equipment important to safety (for example, control room habitability system dampers) would return to its non-safety mode following the reset of an ESF signal.

In addition, many utilities [illegible] safety, [illegible] initiate containment isolation.

SAFETY SIGNIFICANCE

The overriding of certain containment ventilation isolation signals could also bypass other safety actuation signals and thus prevent valve closure when the other isolation signals are present. Although such designs may be acceptable, and even necessary, to accomplish certain reactor functions, they are generally unacceptable where they result in the unnecessary bypassing of safety actuation signals. Where such bypassing is also inadvertent, a more serious situation is created especially where there is no bypass indication system to alert the operator.

Where the resetting of ESF actuation signals, such as safety injection, directly causes equipment important to safety to return to its non-safety mode, protective actions of the affected systems could be prematurely negated when the associated actuation signal is reset. Prompt operator action would be required to assure that the necessary equipment is returned to its emergency mode.

The use of non-safety grade monitor to initiate containment isolation could seriously degrade the reliability of the isolation system.

STAFF POSITION

It is our position that, in addition to other applicable criteria, the following should be satisfied for all operating license applications currently under review:

1) The overriding[a] of any type of safety actuation signal (e.g., particulate radiation) should not cause [illegible] of safety actuation signal (e.g., iodine radiation, chamber pressure) for those valves that have no function other than containment isolation.

2) Physical features (e.g., key lock switches) should be provided to ensure adequate administrative controls.

3) A system level annunciation of the overridden status should be provided for every safety system impacted when any override is active. (See R.G. 1.47).

4) The following diverse signals should be provided to initiate isolation of the containment purge/ventilation system: containment high radiation, safety injection actuation, and containment high pressure (where containment high pressure is not a portion of safety injection actuation).

5) The instrumentation systems provided to initiate containment purge ventilation isolation should be designed and qualified to Class 1E criteria.

6) The overriding or resetting[b] of the ESF actuation signal should not cause any equipment to change position.

Accordingly, you are requested to review your protection system design to determine its degree of conformance to these criteria. You should report the results of your review to us describing any departures from the criteria and the corrective actions to be implemented. Design departures for which no corrective action is planned should be justified.

The following definitions are given for clarity.

[a]Override: The signal is still present, and it is blocked in order to perform a function contrary to the signal.

[b]Reset: The signal has come and gone, and the circuit is being cleared in order to return it to the normal condition.

223.89 (9.5.5) The diesel generators are required to start automatically on loss of all offsite power and in the event of a LOCA. The diesel generator sets should be capable of operation at less than full load for extended periods without degradation of performance or reliability. Should a LOCA occur with availability of offsite power, discuss the design provisions and other parameters that have been considered in the selection of the diesel generators to enable them to run unloaded (on standby) for extended periods without degradation of engine performance or reliability. Expand your PSAR/FSAR to include and explicitly define the capability of your design with regard to this requirement. (SRP 9.5.5, Part III, Item 7).

223.90 Provide a response to OIE Bulletin 79-77, dated November 11, 1979, for Shoreham.

LEVEL MEASUREMENT ERRORS DUE TO ENVIRONMENTAL TEMPERATURE EFFECTS ON LEVEL INSTRUMENT REFERENCE LEGS

223.91 On June 22, 1979, Westinghouse Electric Corporation reported to NRC, a potential safety hazard under 10 CFR 21. This report addresses errors generated in the steam generator level indication sensors following high energy pipe break accidents inside containment. Further, the report implies that previous analyses of peak containment temperature and pressure may have been non-conservative. Breaks of this type can result in heatup of the steam generator level measurement reference leg resulting in a decrease of the water column density with a consequent increase in the indicated steam generator water level (i.e., indicated level exceeding actual level). IE Bulletin 79-21 includes further information on this problem and addresses appropriate actions which are to be taken by licensees of operating plants.

Applicants for an operating license are requested to submit a response to the following questions and to revise their safety analysis report consistent with this response.

1. Describe the liquid level measuring systems within containment that are used to initiate safety actions or are used to provide post-accident monitoring information. Provide a description of the type of reference leg used i.e., open column or sealed reference leg.

2. Provide an evaluation of the effect of post-accident ambient temperatures on the indicated water level to determine the change in indicated level relative to actual water level. This evaluation must include other sources of error including the effects of varying fluid pressure and flashing of reference leg to steam on the water level measurements.

3. Provide an analysis of the impact that the level measurement errors in control and protection systems (2 above) have on the assumptions used in the plant transient and accident analysis. This should include a review of all safety and control setpoints derived from level signals to verify that the setpoints will initiate the action required by the plant safety analyses throughout the range of ambient temperatures encountered by the instrumentation, including accident temperatures. If this analysis demonstrates that level measurement errors are greater than assumed in the safety analysis, address the corrective action to be taken. The corrective actions considered should include design changes that could be made to ensure that containment temperature effects are automatically accounted for. These measures may include setpoint changes as an acceptable corrective action for the short term. However, some form of temperature compensation or modification to eliminate or reduce temperature errors should be investigated as a long term solution.

4. Review and indicate the required revisions, as necessary, of emergency procedures to include specific information obtained from the review and evaluation of Items 1, 2, and 3 to ensure that the operators are instructed on the potential for and magnitude of erroneous level signals. Provide a copy of tables, curves, or correction factors that would be applied to post-accident monitoring systems that will be used by plant operators.