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

Direct Dial Number

November 19, 1982 SNRC-797

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

Mechanical Equipment Environmental Qualification
SER Outstanding Issue No. 9
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Dear Mr. Denton:

Enclosed are forty (40) copies of the justifications for interim operation with safety related mechanical equipment for which outstanding items were identified by the review program described in letter SNRC-737 dated July 23, 1982 from LILCO (J. L. Smith) to the NRC (H. R. Denton). This submittal fulfills the LILCO commitment made in letter SNRC-767 dated September 9, 1982 from LILCO (J. L. Smith) to the NRC (H. R. Denton), Exhibit 3, Question 1. An Index of Submittal has been attached for your convenience.
These justifications for interim operation provide assurance that the Shoreham Nuclear Power Station can be safely operated pending completion of the mechanical equipment environmental qualification program.

If you have any questions regarding this matter, please contact this office.

Very truly yours,

J. L. Smith

J. L. Smith
Manager, Special Projects
Shoreham Nuclear Power Station

JFE:mp

Enclosures

cc: J. Higgins
All parties

A048
s
1/40

INDEX OF SUBMITTAL

Justifications for Interim Operation of
Safety-Related Mechanical Equipment

Mark No.	Equipment Name
1B21*AOV81A-D	Main Steam Inboard Isolation Valves
1C11-AOV126 1C11-AOV127	Scram Inlet and Outlet Valves
1C11-HCU01	Hydraulic Control Unit Scram Accumulator
1C51*EV801A-D	Explosive Shear Valve
1T46*AOV038A 1T46*AOV039A-B	Air Operated Butterfly Valves
VCS-60X-2	Forged Stainless Steel Piston Check Valves

EQUIPMENT INTERIM JUSTIFICATION

Mark No.

1B21*AOV081A,B,C,D

System Name:

Main Steam System

Equipment Name:

Main Steam Inboard Isolation Valve

Manufacturer:

Rockwell International

Model No.:

1612

1.0 SYSTEM AND EQUIPMENT FUNCTION

1.1 System Function

The main steam isolation valves form part of the nuclear system process barrier for openings outside the primary containment, and part of the pressure barrier for nuclear system breaks inside the primary containment.

1.2 Equipment Function

The equipment functions that are to be performed by the main steam isolation valves, when exposed to a LOCA or a PBOC:

a. Main Steam and Containment Isolation for a LOCA or Main Steam Isolation for PBOC.

b. Close the main steam lines within the time established by design basis accident analysis to limit the release of reactor coolant.

The Inboard Main Steam Isolation Valves will experience the postulated harsh environmental conditions resulting from a LOCA during which they must complete the above functions. Additionally, these valves must not fail in a manner detrimental to plant safety or accident mitigation subsequent to a LOCA or PBOC accident.

2.0 NON-METALLIC SUBCOMPONENT(S) REQUIRING INTERIM JUSTIFICATION

2.1 Identification of Subcomponent(s)

The non-metallic subcomponents requiring justification are made of viton. The parameter requiring justification is radiation.

The valve pneumatic and hydraulic control units contain viton seals whose function is to contain the air and hydraulic oil necessary for valve operation.

2.2 Comparison of Postulated Environment and Documented Environment

The dose to the inboard MSIV valves, if required to withstand 40-year normal plus accident radiation, would reach 2.7×10^7 rads. This accident dose is based on a required operating time plus margin of 70 minutes. The radiation tolerance of the viton seals is 1×10^7.

3.0 JUSTIFICATION STATEMENT

As stated above, the 40-year normal plus accident integrated radiation dose would reach 2.7×10^7 which is in excess of the radiation tolerance of viton (1×10^7). However, the 2-year normal radiation plus 70-minute accident integrated dose is 9.8×10^6 rads. Since viton is the limiting material contained within this device and its radiation tolerance is greater than the above 2-year normal radiation plus accident dose, these valves are expected to perform their safety function for at least an interim period of 2 years.

The valves, if they fail, do so in a closed position and are not required to reopen or reshut for the duration of the accident. Furthermore, the outboard MSIVs are qualified for the 40-year plus accident integrated radiation dose.

Based on these considerations, interim plant operation is justified.

EQUIPMENT INTERIM JUSTIFICATION

Mark No.:

1C11-AOV126
1C11-AOV127

System Name:

Control Rod Drive

Equipment Name:

Scram Inlet and Outlet Valves

Manufacturer:

Robertshaw

Model No.:

88470-A1
83460-B2

1.0 SYSTEM AND EQUIPMENT FUNCTION

1.1 System Function

The scram inlet and outlet valves are important for the operation of the Control Rod Drive Hydraulic System. During normal operation, the inlet and outlet scram valves are held closed by control air pressure supplied to the tops of the diaphragm actuators by the scram pilot valves. Upon the receipt of a scram signal, the scram inlet valve opens to supply pressurized water to the bottom of the drive piston. The scram exhaust valve opens slightly before the scram inlet valves, exhausting water from above the drive piston. The differential pressure across the drive piston causes the control rod to insert.

1.2 Equipment Function

The inlet and outlet scram valves are quick opening globe valves operated by an internal spring and system pressure. The scram outlet valve opens slightly before the inlet valve because of a larger spring in the valve operator.

The inlet and outlet scram valves are an essential part of the Hydraulic Control Unit. It is an extremely rapid operating system, fully activating within 4 seconds of receiving a scram signal and 6 seconds following the most limiting accident.

2.0 NON-METALLIC SUBCOMPONENT(S) REQUIRING INTERIM JUSTIFICATION

2.1 Identification of Subcomponent(s)

The non-metallic subcomponents requiring justification are made of teflon. They are the packing set for both valves and the seat ring for the 1C11-AOV127. The parameter requiring justification is radiation.

2.2 Comparison of Postulated Environment and Documented Environment

The inlet and outlet scram valves are exposed to a postulated 40-year normal and 6-month accident integrated radiation dose of 5.75×10^6 rads. The radiation tolerance of the "weak link" subcomponent teflon is 3.4×10^4 rads.

3.0 JUSTIFICATION STATEMENT

The inlet and outlet scram valves are expected to perform their safety function at least up to the radiation tolerance of the weak link (teflon), which as stated above, is 3.4×10^4 rads. This value is less than the postulated value for the 40-year normal and accident integrated dose. However, the scram valves are fast acting and they fully perform their required safety function within 6 seconds following an accident. This activation time is far within the margin of 12 minutes (0.2 hour), which is the acceptable span of time in which the scram valves can perform their function without being affected by radiation. The postulated dose to the scram valves from 40-year normal and 12-minute accident radiation is 3.22×10^4 rads.

Upon initial operaton of the inlet and outlet scram valves, the control rods will be fully inserted and latched secure within 6 seconds following an accident. Thus, any subsequent failure of the scram valves is not detrimental to plant safety or accident mitigation.

Therefore, since the scram valves need function only once, immediately following accident initiation, and then are subsequently not required, the radiation dose seen by the scram valves is far less than the tolerable dose. Thus, the devices are expected to perform their safety function as required.

Based on these considerations, interim plant operation is justified.

EQUIPMENT INTERIM JUSTIFICATION

Mark No.:

1C11-HCU01

System Name:

Control Rod Drive

Equipment Name:

Hydraulic Control Unit
Scram Accumulator

Manufacturer:

General Electric

Model No.:

921D595G

1.0 SYSTEM AND EQUIPMENT FUNCTION

1.1 System Function

The purpose of the Hydraulic Control Unit Scram Accumulator is to provide a backup drive source for the control rods in the event that reactor pressure is low and/or the Control Rod Drive Hydraulic System (CRDHS) fails.

1.2 Equipment Function

The scram accumulator is an hydraulic cylinder with a free floating piston. The piston separates the water on top from the high pressure nitrogen below. Nitrogen is used as the source of stored energy to drive the water in the accumulator. The scram accumulator stores sufficient energy to fully insert a control rod at lower reactor vessel pressures. Once inserted, the control rod is restricted from withdrawing by a latch which holds the rod in place. At higher reactor vessel pressures, the accumulator pressure is assisted by reactor vessel pressure. There are a total of 137 accumulators (one for each control rod).

The scram accumulator forms an integral part of the Hydraulic Control Unit. It is an extremely rapid operating system, fully activating within 4 seconds of receiving a scram signal and 6 seconds following the most limiting accident.

2.0 NON-METALLIC SUBCOMPONENT(S) REQUIRING INTERIM JUSTIFICATION

2.1 Identification of Subcomponent(s)

The non-metallic subcomponents requiring justification are made of teflon. They are the wiper rings and the O-ring seal. The parameter requiring justification is radiation.

2.2 Comparison of Postulated Environment and Documented Environment

The accumulator is exposed to a postulated 40-year normal and 6-month accident integrated radiation dose of 5.75×10^6 rads. The radiation tolerance of the "weak link" subcomponent teflon is 3.4×10^4 rads.

3.0 JUSTIFICATION STATEMENT

The accumulator is expected to perform its safety function at least up to the radiation tolerance of the weak link (teflon), which as stated above, is 3.4×10^4 rads. This value is less than the postulated value for the 40-year normal and accident integrated dose. However, the accumulator is a fast acting system and it fully performs its required safety function within 6 seconds following an accident. This is far within the margin of 12 minutes (0.2 hour), which is the acceptable span of time in which the accumulator can perform its function without being affected by radiation. The postulated dose to the accumulator from 40-year normal and 12-minute accident radiation is 3.22×10^4 rads.

Once the accumulators have functioned, the control rods will be fully inserted and latched secure within 6 seconds following an accident. Thus, any subsequent failure of the accumulators is not detrimental to plant safety or accident mitigation.

Therefore, the device is expected to perform its safety function as required.

Based on these considerations, interim plant operation is justified.

EQUIPMENT INTERIM JUSTIFICATION

Mark No.:

1C51*EV801A, B, C, D

System Name:

Traversing In-Core Probe (TIP) System

Equipment Name:

Explosive Shear Valve

Manufacturer:

Consolidated Controls Corp.

Model No.:

73074

1.0 SYSTEM AND EQUIPMENT FUNCTION

1.1 System Function

The purpose of the Traversing In-Core Probe (TIP) System is to provide a highly accurate flux signal used to calibrate the Local Power Range Monitoring (LPRM). While in service, it provides continuous line plots of the axial flux distribution at 31 locations in the core. The TIP equipment is designed to maintain the integrity of the primary containment if isolation is required.

1.2 Equipment Function

The explosive shear valve is manually activated in an emergency situation. It is used only if containment isolation is required when the TIP is beyond the ball valve and power to the TIP system fails.

The shear valve is located on the detector guide tube between the ball valve and the drive mechanism. The shear valve is a safety device designed to cut the guide tube and drive cable and to seal the guide tube if a leak develops in the reactor coolant system when the drive cable is traversing the guide tube. An explosive charge, detonated by an electric current that is controlled by a key switch mounted on the control panel, provides the force to drive a chisel-shaped slug through the guide tube and drive cable, thereby cutting the cable and sealing the reactor end of the guide tube.

2.0 NON-METALLIC SUBCOMPONENT(S) REQUIRING INTERIM JUSTIFICATION

2.1 Identification of Subcomponents

The non-metallic subcomponents requiring justification are made of teflon. They are the seals, packing material, insert (guillotine seal), and teflon-coated electrical wire. The parameter requiring justification is radiation.

2.2 Comparison of Postulated Environment and Documented Environment

The shear valve is exposed to a postulated 40-year normal and 6 month accident integrated radiation dose of 5.26×10^4 rads. The radiation tolerance of the "weak link" subcomponent teflon is 3.4×10^4 rads.

3.0 JUSTIFICATION STATEMENT

The TIP shear valve is expected to perform its safety function up to at least the radiation tolerance of the weak link (teflon), which, as stated above, is 3.4×10^4 rads. Thus, the integrity of the reactor coolant pressure boundary is assured up to this radiation dose. In comparison, the postulated 40-year normal and 6-month accident integrated radiation dose (5.26×10^4 rads) is slightly greater than the radiation tolerance of the weak link. Thus, it is doubtful whether any significant degradation will occur in the seals. Additionally, several seals must be penetrated in order for the reactor coolant to pass the boundary.

If a containment isolation signal is present while the TIP system is in use (detector is inserted), an automatic withdrawal of the detector drive cable would occur and the ball valve would close to complete the containment isolation function. With a concurrent loss of 120V and 208V AC power to the TIP system, the ball valve, if open, would close and a jam of the drive cable will result. Therefore, the shear valve would be manually activated only if a containment isolation signal is present and the detector cannot be withdrawn.

In addition, the probability of a LOCA in a location that will subject the TIP valve to radiation combined with a loss of power during the time in which the TIP system is operating (plant procedures require the TIP system to operate for only four hours during a month) is quite small.

Thus, it is very unlikely that the integrity of the reactor coolant pressure boundary will be violated due to radiation-induced failure of the TIP shear valve.

Based on these considerations, interim plant operation is justified.

EQUIPMENT INTERIM JUSTIFICATION

PREPARED BY: S. POLLER/
W. DROOKS

MARK NO. 1T46*AOV038A
1T46*AOV039A
1T46*AOV039B*

EQUIPMENT NAME: AIR OPERATED BUTTERFLY VALVE

SYSTEM NAME: REACTOR BUILDING STANDBY VENTILATION SYSTEMS (RBSVS)

MANUFACTURER: FISHER CONTROLS

MODEL NO.: 9220

1. Equipment Function

The RBSVS is initiated during an accident or abnormal condition. When the RBSVS is initiated, the reactor building secondary containment is automatically isolated from the primary containment by the closing of several valves. Valve 1T46*AOV038A is a primary containment purge isolation supply valve. Valve 1T46*AOV039A is a primary containment purge isolation exhaust valve. Both valves are part of the primary containment purge system and are normally closed and not required to operate during normal plant operation or accident conditions. These valves perform no other safety function during an accident than to remain closed for the duration of the accident.

2. Non-Metallic Sub-Components Requiring Interim Justification

T-Ring (EPT-Ethylene/Propylene Terpolymer, Sulphur Cured)

Radiation Threshold = 8.77×10^6 Rads
Maximum Service Temperature = 300° F

An adjustable T-Ring seat is contained in the valve disc. The adjusting set screws and compression ring force the T-ring against the body bore seating surface, which provides a bubble tight shut-off feature.

*This valve has been found to be acceptable based on S&W calculation No. SNPS-1-URB-24-T.

These two purge isolation valves are located within the Drywell where the postulated 40 year normal plus 6 month harsh environment radiation levels are 1.2×10^8 rads for valve IT46*AOV038A and 1.8×10^8 rads for valve IT46*AOV039A and the postulated maximum temperature for both valves is 340°F. These radiation levels and the maximum temperature exceed the radiation threshold level and the maximum service temperature of the EPT material.

3. Justification Statement

Valve IT46AOV038A

This valve is the inboard isolation valve of the inboard and outboard isolation valve combination of IT46*AOV038A & B. The outboard isolation valve (IT46*AOV038B) is located in zone H-10 where the postulated 40 year normal plus 6 month harsh environment radiation level of 5.3×10^4 rads and the postulated maximum temperature of 177°F are well below the threshold radiation level and maximum service temperature of the EPT material. The outboard isolation valve will therefore withstand the harsh environment and will perform its safety function throughout the accident period. Based on the fact that the outboard isolation valve (IT46*AOV038B) is a redundant acceptable component, the required safety function will be accomplished and thus interim operation is justified.

Valve IT46*AOV039A

This valve is the inboard isolation valve of the inboard and outboard isolation valve combination of IT46*AOV039A & B. The outboard isolation valve (IT46*AOV039B) is located in an area of zone K-15 where the postulated 40 year normal plus 6 month harsh environment radiation level of 1.8×10^6 rads and the postulated maximum temperature of 158°F are well below the threshold radiation level and maximum service temperature of the EPT material. The outboard isolation valve will therefore withstand the harsh environment and will perform its safety function throughout the accident period. Based on the fact that the outboard isolation valve (IT46*AOV039B) is a redundant acceptable component, the required safety function will be accomplished and thus interim operation is justified.

EQUIPMENT INTERIM JUSTIFICATION

MARK NO. VCS-60X-2

PREPARED BY: K. MENON

EQUIPMENT NAME: FORGED STAINLESS STEEL PISTON CHECK VALVE

SYSTEM NAME: NUCLEAR BOILER (1B21) INSTRUMENT & SERVICE AIR (1P50)

MANUFACTURER: VELAN ENGINEERING

MODEL NO: WO-203B-13MN

General: Piston check valves of description number VCS-60X-2 located in the primary containment, associated with the Safety Relief Valve accumulators and containment isolation function are addressed in this interim justification.

1. Equipment Function:

   A) Eleven 3/4" piston check valves are used in the Nuclear Boiler System (1B21) as check valves to prevent backflow and loss of pressure in the nitrogen supply lines to each of the accumulators for the short term operation of the individual SRV's.

   B) Four (two 1½" and two 3/4") piston check valves are associated with the containment isolation function of the N_2 supply system.

2. Non-Metalic Sub-Components Requiring Interim Justification

   Soft Ring, Neoprene Disc Seat
   Radiation Threshold = $8x10^5$ rads
   Maximum Service Temp. = 194^oF

   These check valves are all located within the Primary Containment where the postulated 40 year normal plus 180 day LOCA radiation levels range from $1.0x10^8$ rads to $1.8x10^8$ rads and the postulated maximum temperatures range from 225^oF to 340^oF. The postulated radiation levels and the maximum temperatures exceed the radiation threshold level and the maximum service temperature of the neoprene material.

   Exposure to this postulated harsh environment may result in the deterioration of neoprene which could potentially cause these valves to lose their sealing characteristics.

3. Justification Statement

   A) Safety Relief Valves:

   There are four modes of supplying nitrogen to the SRV's (normal operation, short term supply, intermediate term supply, and long term supply).

   During normal operation, nitrogen is supplied to the short term and intermediate term accumulators. Nitrogen is retained in these accumulators by check valves so they remain pressurized and are available for operation of SRV's in case the normal nitrogen supply fails during normal operation.

In the event of an accident the nitrogen supply header that normally supplies nitrogen to the drywell SRV accumulator headers is automatically isolated by a LOCA signal by environmentally qualified motor operated valves (1P50*MOV113A&B) located outside containment.

In addition, two other environmentally qualified motor operated valves (1P50*MOV105A&B), located inside the drywell, are automatically closed to divide the supply header piping. The closing of these additional valves also isolates other lines from the now divided accumulator supply headers inside the drywell. Isolation of the piping system in this manner will eliminate the possibility of pressure loss in the accumulators caused by any backflow leakage due to deteriorated check valve neoprene seats.

Intermediate term supply to the SRV accumulators can be accomplished by two intermediate term nitrogen accumulators each capable of supplying sufficient nitrogen for a minimum of 48 hours of operation. This backup of intermediate nitrogen supply to the divided header is made available by automatic opening (on a LOCA signal) of environmentally qualified motor operated valves (1P50*MOV114A&B) located in the connecting piping downstream of valves 1P50*MOV113A&B that have isolated the N_2 normal supply header. This backup supply of nitrogen further ensures that the nitrogen headers serving the SRV's remains pressurized and thus prevents any backflow out of the short term accumulators resulting from any deterioration of the neoprene seats of the piston check valves.

Beyond 48 hours, for long term operation of the SRV's, the pneumatic system uses connections located outside the reactor building to sustain SRV operability indefinitely.

Based upon the redundant supply of nitrogen to the SRV accumulators, the required safety function will be accomplished and thus interim operation is justified.

B) Containment Isolation Function of VCS-60X-2:

Drywell - Two 1½" piston check valves located on the nitrogen supply header inside the drywell function, on reverse flow, as containment isolation valves.

In the event of an accident with the postulated harsh environment, the neoprene valve seat may deteriorate which could cause these valves to lose their sealing characteristics. If a subsequent loss of nitrogen supply to the drywell occurs, * these piston check valves may experience backflow leakage out of the nitrogen supply header. This loss of nitrogen would be detected by environmentally qualified pressure transmitters 1P50*PT116A&B and environmentally justified pressure switches 1P50*PS105A&B. At that time an environmentally qualified valve, either 1P50IMOV103A or B, could be remote manually closed to provide containment isolation.

* As previously stated in the SRV justification, the possibility of loss of pressure in the accumulators and associated nitrogen supply header piping is extremely remote.

Wetwell - Two 3/4" piston check valves located on the nitrogen supply header inside the wetwell function, on reverse flow, as containment isolation valves.

The supply header in the wetwell is used only to test the downcomer vacuum breaker valves. In the event of a LOCA signal, environmentally qualified motor operated valves (1P50*MOV104 and 1P50*MOV106) would close to isolate the containment. Therefore, even if the neoprene valve seats were to deteriorate because of the postulated harsh environment, containment isolation would not be jeopardized. Furthermore, these environmentally qualified motor operated valves are normally closed, eliminating the possibility of containment leakage.

Based on the redundancy of the environmentally qualified motor operated isolation valves, the required safety function will be accomplished and thus interim operation is justified.