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Address Reply to: Post Office Box 767
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October 22, 1984

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

Subject: LaSalle County Station Units 1 and 2
Request for Additional Information
Regarding TMI Action Item PLan II.D.1
NRC Docket Nos. 50-373 and 50-374

Reference (a): Letter dated July 24, 1984 from A. Schwencer to D. L. Farrar.

Dear Mr. Denton:

Attached is Commonwealth Edison's response to the referenced letter concerning performance testing of BWR Safety/Relief valves. It is our judgment that the attached information adequately demonstrates the applicability of the BWR Owners Group Test Report (NEDE-24988-D) to LaSalle County Station.

Please direct any questions you may have regarding this matter to this office.

One signed original and fifteen copies of this letter are provided for you use.

Very truly yours,

J G Marshall

J. G. Marshall
Nuclear Licensing Administrator

lm

cc: LaSalle Resident Inspector
A. Bournia

Attachment

A046
1/1

9364N

NRC QUESTION 1

The BWR/GE test program utilized a "rams head" discharge pipe configuration. Most plants utilize a "tee" quencher configuration at the end of the discharge line. Describe the discharge pipe configuration used at your plant and compare the anticipated loads in this configuration to the measured loads in the test program. Discuss the impact of any differences in loads on valve operability.

RESPONSE TO QUESTION 1

The safety/relief valve discharge piping configuration at LaSalle County Station Units 1 and 2 utilizes a "tee" quencher at the discharge pipe exit. The average length of the 18 SRV discharge lines (SRVDL) leading to the suppression pool is 157 ft. and the submergence length in the suppression pool is approximately 23 ft. The SRV test program utilized a rams head at the discharge pipe exit, a pipe length of 112 ft. and a submergence length of approximately 13 ft. Loads on valve internals during the test program are larger than loads on valve internals in the LaSalle County Station Units 1 and 2 configuration for the following reasons:

1. No dynamic mechanical load originating at the "tee" quencher is transmitted to the valve in the LaSalle Station configuration because there is at least one anchor point between the valve and the "tee" quencher.
2. The first length of the segment of piping downstream of the SRV in the test facility was longer than the LaSalle County Station Units 1 and 2 piping, thereby resulting in a bounding dynamic mechanical load on the valve in the test program. The first segment length in the test facility is 12 ft. whereas this length is a maximum of 8'-8" in the plant configuration.
3. Dynamic hydraulic loads (backpressure) are experienced by the valve internals in the LaSalle Station configuration. The backpressure loads may be either (i) transient backpressures occurring during valve actuation, or (ii) steady-state

backpressures occurring during steady-state flow following valve actuation.

(a) The key parameters affecting the transient backpressures are the fluid pressure upstream of the valve, the valve opening time, the fluid inertia in the submerged SRVDL and the SRVDL air volume. Transient backpressures increase with higher upstream pressure, shorter valve opening times and greater line submergence, and decrease with greater SRVDL air volume. The maximum transient backpressure occurs with high pressure steam flow conditions - a condition that LaSalle County Station Units 1 and 2 have experienced during operation. Furthermore, an in-plant SRV test was performed on LaSalle County Station Unit 1 to demonstrate that the design criteria bound the actual loads on the SRV discharge lines and containment. The transient backpressure for the alternate shutdown cooling mode of operation is always much less than that for the design for steam flow conditions because of the lower upstream pressure and the slower valve opening time.

(b) The steady-state backpressure in the test program was maximized by utilizing an orifice plate in the SRVDL above the water level and before the ramshead. The orifice was sized to produce a backpressure greater than that calculated for any of the LaSalle County Station Units 1 and 2 SRVDLs.

Because of the differences in the line configuration between the LaSalle County Station Units 1 and 2 and the test program, as discussed above, the resultant loads on the valve internals for the test facility bound the actual LaSalle Station loads. An additional consideration in the selection of the ramshead for the test facility was to allow more direct measurement of the thrust load in the final pipe segment. Utilization of a "tee" quencher in the test program would have required quencher supports that would unnecessarily obscure accurate measurement of the pipe

thrust loads. For the reasons stated above, differences between the SRVDL configurations at LaSalle Station and the test facility result in more severe loads during the tests; therefore, SRV operability at LaSalle Station is confirmed by the tests.

NRC QUESTION 2

The test configuration utilized no spring hangers as pipe supports. Plant specific configurations do use spring hangers in conjunction with snubber and rigid supports. Describe the safety relief valve pipe supports used at your plant and compare the anticipated loads on valve internals for the plant pipe supports to the measured loads in the test program. Describe the impact of any differences in loads on valve operability.

RESPONSE TO QUESTION 2

The LaSalle County Station Units 1 and 2 safety/relief valve discharge lines (SRVDLs) are supported by a combination of snubbers, rigid supports, and spring hangers. The locations of snubbers and rigid supports at LaSalle Station are such that the location of such supports in the BWR generic test facility is prototypical, i.e., in each case (at LaSalle and at the test facility) there are supports near each change of direction in the pipe routing. Additionally, each SRVDL at LaSalle Station has only 1 or 2 spring hangers, all of which are located in the drywell. The spring hangers, snubbers and rigid supports were designed to accommodate combinations of loads resulting from piping dead weight, thermal conditions, seismic and suppression pool hydrodynamic events, and a high pressure steam discharge transient during a steam discharge event.

The dynamic load effects on the piping and supports of the test facility due to the water discharge event (the alternate shutdown cooling mode) were found to be significantly lower than corresponding loads resulting from the high pressure steam discharge event. As stated in NEDE-24988-P, this finding is considered generic to all BWRs since the test facility was designed to be prototypical of the features pertinent to this issue.

During the water discharge transient there will be significantly lower dynamic loads resulting from the valve operation and subsequent water flow acting on the snubbers and rigid supports than during the steam discharge transient. This more than offsets the small increase in the dead load on these supports due to the weight of the water during the alternate shutdown cooling mode of operation. Therefore, design adequacy of the snubbers and rigid supports is assured as they are designed for the larger steam discharge transient loads.

This question addresses the design adequacy of the spring hangers with respect to the increased dead load due to the weight of the water during the liquid discharge transient. As was discussed with respect to snubbers and rigid supports, the dynamic loads resulting from liquid discharge during the alternate shutdown cooling mode of operation are significantly lower than those from the high pressure steam discharge. Therefore, sufficient margin exists in the LaSalle Station piping system design to adequately offset the increased dead load on the spring hangers in an unpinned condition due to a water filled condition. Furthermore, the effect of the water deadweight load does not affect the ability of SRVs to open and to establish the alternate shutdown cooling path because the loads occur in the SRVDL only after valve opening.

NRC QUESTION 3

Report NEDE-24988-P did not identify any valve functional deficiencies or anomalies encountered during the test program. Describe the impact of valve safety function of any valve functional deficiencies or anomalies encountered during the program.

RESPONSE TO QUESTION 3

No functional deficiencies or anomalies of the safety relief or relief valves, were experienced during the testing at Wyle Laboratories for compliance with the alternate shutdown cooling mode requirement. All the valves subjected to test runs, valid and invalid, opened and closed without loss of pressure integrity or

damage. Anomalies encountered during the test program were all due to failures of test facility instrumentation, equipment, data acquisition equipment, or deviation from the approved test procedure.

The test specification for each valve required six runs. Under the test procedure, any anomaly caused the test run to be judged invalid. No anomalies were reported in the test report. The Wyle Laboratories test log sheet for the Crosby valve tests are attached. These valves are used at the LaSalle Station. No anomalies are reported for the Crosby 6R10 valve tested.

Each Wyle test report for the respective valves identifies each test run performed and documents whether or not the test run is valid or invalid, and states the reason for considering the run invalid. No anomaly encountered during the required test program affects any valve safety or operability function.

All valid test runs are identified in Table 2.2-1 of NEDE-24988-P. The data presented in Table 4.2-1 for each valve were obtained from the Table 2.2-1 test runs and were based upon the selection criteria of:

(a) Presenting the maximum representative loading information obtained from the steam run data,

(b) Presenting the maximum representative water loading information obtained from the 15°F subcooled water test data,

(c) Presenting the data on the only test run performed for the 50°F subcooled water test condition.

NRC QUESTION 4

The purpose of the test program was to determine valve performance under conditions anticipated to be encountered in the

plants. Describe the events and anticipated conditions at your plant for which the valves are required to operate and compare these plant conditions to the conditions in the test program. Describe the plant features assumed in the event evaluations used to scope the test program and compare them to plant features at your plant. For example, describe high level trips to prevent water from entering the steam lines under high pressure operating conditions as assumed in the test event and compare them to trips used at your plant.

RESPONSE TO NRC QUESTION 4

The purpose of the S/RV test program was to demonstrate that the Safety/Relief Valves (S/RVs) will open and reclose under all expected flow conditions. The expected valve operating conditions were determined through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2 and described in FSAR Chapter 15. Single failures were assumed for these analyses so that the dynamic forces on the safety and relief valves would be maximized. Test pressures were the highest predicted by conservative safety analysis procedures. The BWR Owners Group, in their enclosure to the September 17, 1980 letter from D. B. Waters to R. H. Vollmer, identified 13 cases which may result in liquid or two-phase S/RV inlet flow that would maximize the dynamic forces on the safety/relief valves. These cases were identified from an evaluation of the initial events described in Regulatory Guide 1.70, Revision 2, with and without the additional conservatism of a single active component failure or postulated operator error in the event sequence. It was concluded from this evaluation that the alternate shutdown cooling mode is the only case which can result in liquid or two-phase fluid at the valve inlet. Consequently, this was the case simulated in the S/RV test program. This conclusion and the test results applicable to LaSalle County Station Units 1 and 2 are discussed below. The alternate shutdown cooling mode of operation is described in the response to NRC Question 5.

The S/RV inlet fluid conditions tested in the BWR Owners Group S/RV test program, as documented in NEDE-24988-P, are 15°F to

50°F subcooled liquid at 20 psid to 250 psid. These fluid conditions envelope the conditions expected to occur at LaSalle County Station Units 1 and 2 in the alternate shutdown cooling mode for operation.

The BWR Owners Group identified 13 cases by evaluating the initiating events described in Regulatory Guide 1.70, Revision 2, with the additional conservatism of a single active component failure or operator error postulated in the events sequence. These cases and the plant-specific features that mitigate these events, are summarized in Table 1. Of these 13 cases, only 8 are applicable to LaSalle County Station because of its design and specific plant configuration. Five cases, namely 2, 3, 8, 11, and 13 are not applicable to LaSalle County Station Units 1 and 2 for the reasons listed below:

a. Case 2 - Results in steam flow only because the S/RVs are located higher than the MSIVs.

b. Case 3 - There is no HPCI system at LaSalle County Station Units 1 and 2.

e. Case 8 - Results in steam flow only because the S/RVs are located higher than the MSIVs.

f. Case 11 - There is no HPCI system at LaSalle County Station Units 1 and 2.

h. Case 13 - There are no procedures requiring break isolation. The operator is trained to respond to high water level indication and alarms before the vessel is filled to the MSL level.

For the eight remaining cases, the LaSalle specific features, such as trip logic, power supplies, instrument line configura-

tion, alarms and operator actions, were compared to the base case analysis presented in the BWR Owners Group submittal of September 17, 1980. The comparison demonstrated that for each case, the base case analysis was applicable to LaSalle Units 1 and 2 because the base case analysis does not include any plant features which are not already built into the LaSalle design. For cases 1, 4, 5, 6, 7, 9, 10, and 12, Table 1 shows that LaSalle specific features are included in the base case analysis presented in the BWR Owners Group submittal of September 17, 1980. From Table 1, it is evident that all plant features assumed in the base case evaluation are also included in the LaSalle plant. Furthermore, the time available for operator action at LaSalle Station is expected to be longer than the time interval used in the base case analysis for each case where operator action is required.

Event 7, the alternate shutdown cooling mode of operation, is the only expected event which results in liquid or two-phase fluid at the S/RV inlet. Consequently, this event was simulated in the BWR S/RV test program. For LaSalle Station, this event involves flow of subcooled water (approximately 15°F to 50°F subcooled) at a pressure of approximately 200 psig to 250 psig. The S/RV test conditions clearly enveloped these plant conditions.

As discussed above, the BWR Owners Group evaluated transients including single active failures that would maximize the dynamic forces on the safety/relief valves. As a result of this evaluation, the alternate shutdown cooling mode is the only expected event involving liquid or two-phase flow. Consequently this event was tested in the BWR S/RV test program. The fluid condi tions and flow conditions tested in the BWR Owners Group test program conservatively envelope the LaSalle County Station Units 1 and 2 plant-specific fluid conditions expected for the alternate shutdown cooling mode of operation.

NRC QUESTION 5

The valves are likely to be extensively cycled in a controlled depressurization mode in a plant specific application. Was this mode simulated in the test program? What is the effect of this valve cycling on valve performance and probability of the valve to fail open or to fail close?

RESPONSE TO NRC QUESTION 5

The BWR safety/relief valve (SRV) operability test program was designed to simulate the alternate shutdown cooling mode, which is the only expected liquid or two-phase flow discharge event for LaSalle Station. The sequence of events leading to the alternate shutdown cooling mode is given below.

Following normal reactor shutdown, the reactor operator depressurizes the reactor vessel by opening the turbine bypass valves to dump heat to the main condenser. If the main condenser is unavailable, the operator could depressurize the reactor vessel by using the SRV's to discharge steam into the suppression pool. When depressurization is need. the operator manually cycles the valves to assure that the vessel cooldown rate is maintained with the technical specification limit of 100°F per hour. As soon as the vessel is depressurized, the operator initiates the normal shutdown cooling mode of operation via the RHR system. If that mode is unavailable for any reason, the operator can initiate the alternate shutdown cooling mode which involves the suppression pool and RHR loops.

As discussed in the preceeding paragraph, if the normal equipment is postulated to be unavailable, then the operator will initiate the alternate shutdown cooling mode of operation. For alternate shutdown cooling, the operator opens one or more SRVs and initiates either an RHR or LPCS loop utilizing the suppression pool as the suction source. (The suppression pool at LaSalle has a clean-up loop to maintain reactor quality water.) The reactor vessel is filled such that water flows through open SRV(s) back to the suppression pool. Cooling of the suppression pool is

provided by use of an RHR heat exchanger where essential service water transfers heat to teh Ultimate Heat Sink. This is the alternate cooling mode at LaSalle.

To assure continuous long-term heat removal, an SRV is kept open and no cycling of the valve is performed. In order to control the reactor vessel cooldown rate, the operator can limit flow into the vessel by throttling the RHR (or LPCS) injection valves. By design, no cycling of the SRV is required for the alternate shutdown cooling mode, hence no cycling of the SRV was performed for the generic BWR SRV operability test program.

The ability of the LaSalle SRV's to be extensively cycled for steam discharge conditions has been confirmed during steam discharge qualification testing of the valve by the valve vendor and during start-up testing at LaSalle. Based on the qualification testing of the SRVs, cycling of the valves in a controlled depressurization mode for steam discharge conditions does not adversely affect valve performance and thus the probability of the valve to fail open or closed is extremely low.

NRC QUESTION 6

Describe how the valves of valve C_v's in report NEDE-24988-P will be used at your plant. Show that the methodology used in the test program to determine the valve C_v is consistent with your application.

RESPONSE TO NRC QUESTION 6

See the LaSalle specifc response to FSAR Question 212.46 provided via Amendment 50 (October 1980) and Amendment 56 (May 1981). The flow coefficient, C_v, for the Crosby safety relief valves (SRVs) utilized at LaSalle Station was determined in the generic SRV test program (NEDE-24988-P). The average flow coefficient Crosby valves as calculated from the test results is reported in Table 5.2-1 of NEDE-24988-P. This test value has been used by Commonwealth Edison Company to confirm that the liquid discharge flow capacity of the LaSalle SRVs is sufficient to remove core decay

heat via the alternate shutdown cooling mode. The C_v value determined in the SRV test is sufficient to allow return of the RHR (or LPCS) pump flow to the suppression pool. Eighteen SRV's are provided for each Unit at LaSalle. The calculated flow capacity per valve is 1400 gpm under shutdown conditions.

When using the alternate shutdown cooling mode, the operator can assure that adequate core cooling is being provided by monitoring the following parameters: RHR (or LPCS) flow rate, vessel pressure, vessel level, and reactor coolant temperature and suppression pool temperature and level.

The flow coefficients for the Crosby valves reported in NEDE-24988-P were determined from the SRV flow rate when the valve inlet was pressurized to approximately 250 psig. The valve flow rate was measured with the supply line flow venturi upstream of the steam chest. The C_v for the valve was calculated using the nominal measured pressure differential between the valve inlet (steam chest) and 3 ft. downstream of the valve at the corresponding measured flowrate. Furthermore, these test conditions and test configuration were representative of LaSalle conditions for the alternate shutdown cooling mode, e.g. pressure upstream of the valve, fluid temperature, friction losses and liquid flowrate. Therefore the reported C_v values are appropriate for application to the LaSalle County Station Units 1 and 2 plant.

TABLE 1 - CASES EVALUATED

PLANT FEATURES	# 1 FW Cont. Fail., FW L8 Trip Failure	# 2 Press. Reg. Fail.	# 3 Transient HPCI, HPCI L8 Trip Failure	# 4 Transient RCIC, RCIC L8 Trip Failure	# 5 Transient HPCS, HPCS L8 Trip Failure	# 6 Transient RCIC Hd. Spr.	# 7 Alt. Shutdown Cooling, Shutdown Suction Unavailable	# 8 MSL Brk OSC	# 9 SBA, RCIC, RCIC L8 Trip Failure	#10 SBA, HPCS, HPCS L8 Trip Failure	#11 SBA, HPCI, HPCI L8 Trip Failure	#12 SBA, Depress. & ECCS Over., Operator Error	#13 LBA, ECCS Overf Brk Isol
Low Pressure ECCS Initiation on High Drywell Pressure												X / S	X / NA
Low Pressure Initiation on Low Water Level													X / NA
FW Pumps Trip on Low Suction Pressure	X / S												
HPCS Trip on High Back-pressure			X / NA								X / NA		
RCIC Trip on High Back-pressure				X / S					X / S				
Turbine Trip on Vessel High Level	X / S	X / NA											
MSIVs Closure on Low Turbine Inlet Pressure	X / S	X / NA						X / NA					
MSIVs Closure on High Steam Flow		X / NA						X / NA					
MSIVs Closure on High Steam Tunnel Temperature								X / NA					

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PLANT FEATURES	# 1 FW Cont. Fail., FW L8 Trip Failure	# 2 Press. Reg. Fail.	# 3 Transient HPCI, HPCI L8 Trip Failure	# 4 Transient RCIC, RCIC L8 Trip Failure	# 5 Transient HPCS, HPCS L8 Trip Failure	# 6 Transient RCIC Hd. Spr.	# 7 Alt. Shutdown Cooling, Shutdown Suction Unavailable	# 8 MSL Brk OSC	# 9 SBA, RCIC, RCIC L8 Trip Failure	#10 SBA, HPCS, HPCS L8 Trip Failure	#11 SBA, HPCI, HPCI L8 Trip Failure	#12 SBA, Depress. & ECCS Over., Opèrator Error	#13 LBA, ECCS Overf Brk Isol
High Water Level 7 Alarm	X / S		X / NA	X / S	X / S				X / S	X / S	X / NA	X / S	X / NA
High Drywell Pressure Alarm													
FW Level 8 Trip	X / S	X / NA											
RCIC Level 8 Trip			X / NA	X / S	X / S				X / S	X / NA	X / NA		X / NA
HPCS Level 8 Trip				X / S	X / S				X / S	X / S			X / NA
HPCI Level 8 Trip			X / NA	X / NA					X / NA		X / NA		X / NA
HPCI/S and RCIC Initiation on Low Water Level	X / S	X / NA	X / NA	X / S	X / S	X / S		X / NA	X / S				X / NA
HPCI/S Initiation on High Drywell Pressure			X / NA	X / S					X / S	X / S	X / NA	X / S	X / NA
RCIC Initiation on High Drywell Pressure													X / NA

TABLE 1 - CASES EVALUATED

PLANT FEATURES	# 1 FW Cont. Fail., FW L8 Trip Failure	# 2 Press. Reg. Fail.	# 3 Transient HPCI, HPCI L8 Trip Failure	# 4 Transient RCIC, RCIC L8 Trip Failure	# 5 Transient HPCS, HPCS L8 Trip Failure	# 6 Transient RCIC Hd. Spr.	# 7 Alt. Shutdown Cooling, Shutdown Suction Unavailable	# 8 MSL Brk OSC	# 9 SBA, RCIC, RCIC L8 Trip Failure	#10 SBA, HPCS, HPCS L8 Trip Failure	#11 SBA, HPCI, HPCI L8 Trip Failure	#12 SBA, Depress. & ECCS Over., Operator Error	#13 LBA, ECCS Overf Brk Isol
MSIV Closure on High Radiation								X/NA					
Reactor Scram on Turbine Trip	X/S	X/NA											
Reactor Scram on Neutron Flux Monitor		X/NA											
Reactor Scram on MSIVs Closure		X/NA											
Reactor Scram on High Radiation								X/NA					
Reactor Scram on High Drywell Pressure									X/S	X/S	X/NA	X/S	X/NA
Reactor Scram on Low Water Level													X/NA
Reactor Isolation on Low Water Level													X/NA

KEY: X - Feature considered in Base Case Analysis
S - Feature in Plant Specific Design
NA - Not Applicable

OPERABILITY TEST REPORT

FOR

CROSBY 6R10 SRV

FOR

LOW PRESSURE WATER TESTS

FOR

GENERAL ELECTRIC COMPANY

GENERAL ELECTRIC

NUCLEAR ENERGY BUSINESS GROUP

APPROVED	DATE
3682-78-1	
VPF NO.	
D810271	
TRANSMITTAL NO.	
PRINTS TO	

175 Curtner Avenue
San Jose, California

TABLE I

OPERABILITY TEST LOG, SRV CR-1

TEST NO.	TEST MEDIA	LOAD LINE CONFIGURATION	TEST DATE	REMARKS
401	Steam	I	3/24/81	Backpressure low, changed orifice.
402	Steam	I	3/24/81	Test Acceptable
403	Water	I	3/24/81	Test Acceptable
404	Steam	I	3/24/81	Test Acceptable
405	Water	I	3/25/81	Test Acceptable
406	Steam	I	3/25/81	Test Acceptable
407	Water	I	3/25/81	Test Acceptable