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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 18 1977

MEMORANDUM FOR: D. G. Eisenhut, Assistant Director for Operational Technology, DOR

FROM: R. L. Tedesco, Assistant Director for Plant Systems, DSS

SUBJECT: EVALUATION OF OYSTER CREEK MODIFIED PRESSURE SET-POINT FOR SAFETY/RELIEF VALVE (TAC-2285)

On November 24, 1976, the Oyster Creek Nuclear Generating Station conducted testing of the safety/relief valve (SRV) to discharge system to investigate the hydrodynamic loads on the torus shell and SRV piping and supports. The test results were documented by letter dated January 10, 1977 from J. R. Fintock of Jersey Central Power and Light Company to George Lear. The Containment Systems Branch is currently reviewing this document but has not completed the evaluation at this time because of the short review time. However, we find an urgent need to comment on one particular area of concern, i.e., the action taken by the licensee to change the SRV setpoint.

The report indicates that actions have been taken by the licensee to mitigate SRV loads. The most significant one is changing the SRV pressure setpoint; i.e., the set point of one of the three SRVs connected to a common header was lowered from 1070 psig to 1050 psig while maintaining the other two SRVs at the original 1070 psig setpoint. This action is intended to avoid the case of simultaneous actuation of the three SRVs which the report considers the most severe loading condition. However, based on our evaluation, staggering of these three SRV setpoints could potentially create an even more severe load case by having the two high setpoint SRVs actuated in conditions equivalent to subsequent SRV actuation.

A detailed evaluation is presented in the enclosed report. The following summarizes our evaluation:

1. We have concluded that changing the SRV pressure set-point while maintaining the current SRV discharge system will not improve the situation. Rather, it may potentially aggravate the SRV loads. Therefore, we believe that all SRVs for Oyster Creek Station should be reset to the original setpoint of 1070 psig.

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2. The load factors provided in Table 5-1 of the Jersey Central Power and Light Company Report are acceptable for interim structural evaluation, if the SRV's are reset to 1070 psig. If the current staggering SRV setpoint is maintained, an additional load case, i.e., two valves into a hot pipe, with a load factor of 3.4 should be included in the structural evaluation. Note that these load factors are accepted on an interim basis.

3. If the structural evaluation indicates a need for load mitigation, we would recommend consideration of the following:

   a. each SRV should be channeled to its individual line and individual discharge end;

   b. the SRVs should be distributed evenly within the pool;

   c. use of a ramshead discharge device should be considered. If the current elbow type of discharge end is maintained, in plant test should be performed after the modification indicated in a and b above is conducted.

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Preliminary Evaluation of Oyster Creek Report On
Safety/Relief Valve Tests

On November 24, 1976, the Oyster Creek Nuclear Generating Station conducted the testing of the safety/relief valve (SRV) discharge system to investigate the hydrodynamic loads on the torus shell and SRV piping and supports. Report of the test results was documented by letter dated January 10, 1977 from I. R. Fintrock of Jersey Central Power and Light Company (JC) to George Lear. We have reviewed this report and have the following preliminary comments.

To mitigate the predicted load increase due to multiple-valve actuation, JC has taken action to change the pressure set point of two SRVs. However, based on the information available for us, we believe that changing the SRV pressure setpoint will not reduce the loads; instead, it could potentially increase the loads for the following reasons.

At Oyster Creek, there are a total of five SRVs located at the main steam header, three valves are channeled to the south discharge header, and two valves are located on the north header. The south header will result in more severe loading conditions. Therefore, we will concentrate on the phenomena in the south header.

The south header relief valve piping is shown on Figure 1. It consists of individual 8" line jointed into a common 14-inch header. The 14-inch pipe is reduced to 12-inch as it enters the vent line into the torus.

As provided in the JC report, there are approximately 57.0 feet of 8-inch pipe, 68 feet of 12-inch pipe, and 35 feet of 14-inch pipe in the piping system. Based on this information, we estimate that for valve No. 3 the total air volume from the SRV downstream to the discharge point in the torus is 74 cu. ft. The total air volume for valve Nos. 1 and 2 branch lines is 29 cu. ft. This represents about 40% of the total air volume for the valve No. 3.

The JC report indicates that the pressure set-point for one of the three valves has been reduced from 1070 psig to 1050 psig while the other two valves remain at 1070 psig.

For certain transients which cause the reactor to be pressurized, No. 3 valve, the lower setpoint valve, will be first activated. Mass and energy released from the primary system through this valve pressurizes the entire SRV discharge system. The water leg is accelerated toward the pool. When the water leg is cleared, the high pressure air inside the No. 3 SRV line discharges into the pool and results in air bubble loads on the surrounding structures.

Note that the air volume stored in the Nos. 1 and 2 branch lines has been compressed and remains inside the line during this event. This air volume within the SRV lines will be heated due to the presence of the high temperature steam. Based on the Monticello test results, the line temperature reaches 300°F within a minute.

Let us postulate that Nos. 1 and 2 are now activated. Meantime, the 40% air volume contained in the branch line is at high pressure and high temperature; the RPV line is also at high temperature. These conditions are similar with the initial conditions for subsequent SRV actuations, which results in significant load increase as Monticello tests show. Let us call this case as "two-valves into a hot pipe."

The two valves into a hot pipe case is not included in the report. Four cases have been analyzed and the results are summarized below.

Transient	Load Factor
One Valve, Cold pipe	1.0
One Valve, Hot pipe	2.0
Two Valves, Cold pipe	1.7
Three Valves, Cold pipe	2.5

Based on these results, we estimated that the load factor for two valves into a hot pipe is 3.4 (1.7 x 2.0) e.g., load factor for two valves into cold pipe times load factor for discharge from cold to hot pipe. Comparing with the three valves into a cold pipe, we conclude that staggering the pressure set point results in a potential load increase by 36%. It is obvious that the approach of staggering the pressure set point as a means for mitigating the SRV load is not acceptable.

As a result of our evaluation, we have the following conclusion and recommendations:

1. We have concluded that changing the SRV pressure set-points while maintaining the current SRV discharge system will not improve the situation rather potentially aggravating the SRV loads. Therefore, we recommend that all SRVs for Oyster Creek Station should be reset to their original setpoint of 1070 psig.

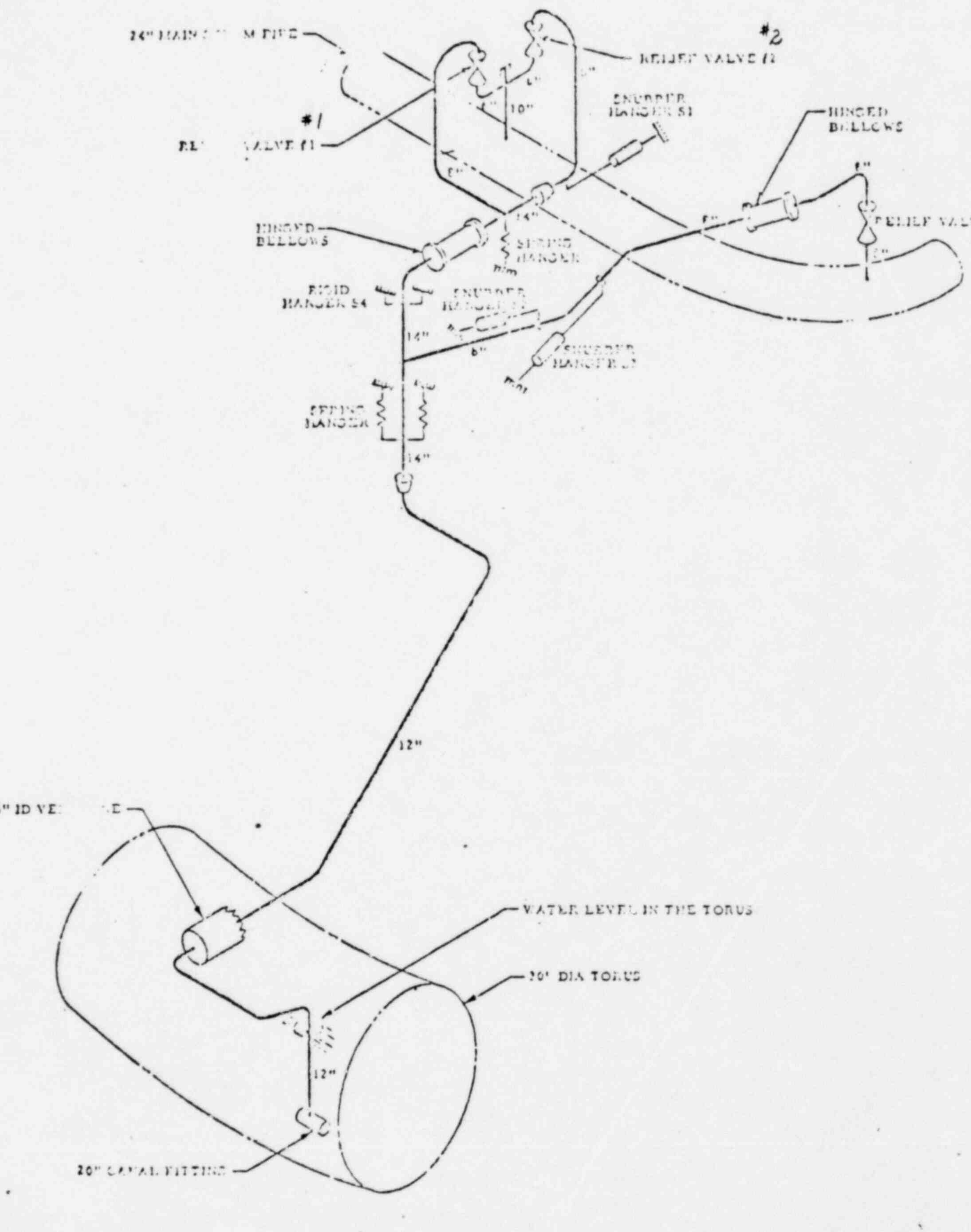
2. The load factors provided in Table 5-1 of the Jersey Central Power and Light Company Report are acceptable for interim structural evaluation, if the action indicated on item 1 above is done. If the current staggering SRV setpoint is maintained, an additional load case, i.e., two valves into a hot pipe, with a load factor of 3.4 should be included in the structural evaluation. Note that these load factors are accepted on an interim basis.

3. If the structural evaluation indicates a need for load mitigation, we would recommend the following:

    a. Each SRV should be channeled to its individual line and individual discharge end;

    b. The SRV discharge ends in pool should be distributed evenly;

c. Usage of ramshead as discharge device should be considered. If the current elbow type of discharge end is maintained, in plant test should be performed after the modification indicated in a and b above is conducted.



SOUTH HEADER RELIEF VALVE DISCHARGE PIPING

FIGURE 1