SAFETY EVALUATION REPORT

OF THE

INTERIM ASSESSMENTS

OF

MULTIPLE-CONSECUTIVE SAFETY-RELIEF

VALVE ACTUATIONS

IN

MARK I CONTAINMENT PLANTS

INTRODUCTION

On October 6, 1977, the General Electric Company (GE) informed the staff of a design deficiency in the safety-relief valve (SRV) control system for the BWR-6 /Mark III system design. In a letter dated October 11, 1977 (G. Sherwood, GE, to N. Mosley, NRC) and during a meeting between representatives from the Mark I Owners' Group, GE, and the NRC staff on October 27, 1977, the implications of this design deficiency to the operating BWR facilities were discussed. Subsequently, the staff requested that each utility formally submit their basis for continued plant operation by November 1, 1977. In letters dated March 20, 1978, the staff presented the assumptions and criteria that were to be used for interim plant-specific assessments of the affected operating BWR facilities, until this issue could ultimately be resolved as part of the Mark I Containment Long-Term Program (LTP).

The design deficiency identified by GE concerns the potential for multiple-consecutive SRV actuations following a reactor isolation transient event. Isolation of the primary system will cause a pressure rise within the reactor vessel. When the pressure reaches the setpoints of the SRVs, the valves will open and discharge steam into the suppression pool, thereby counteracting and eventually reducing the primary system pressure. When the system pressure drops to approximately 800 to 900 psia, the valves will automatically reclose. However, the decay heat produced by by the core will cause the pressure to rise again resulting in repeated SRV actuations. Consecutive SRV actuations, referred to as "hot pops," cause a higher loading on the suppression chamber (torus) and its support structures due to an increase in the length of the water leg and internal energy of the airspace in the SRV discharge line, as compared to that normally existing prior to an SRV actuation. Multiple SRV actuations also cause a higher net loading on the torus due to a reinforcing of multiple sources to specific locations on the torus.

During the Mark I Containment Short-Term Program (STP), operating experience and in-plant test data from the Quad Cities plant indicated that the loads were sufficiently low that only fatigue cycling need be considered. On this basis, SRV discharge loads were classified as secondary loads and excluded from a more detailed consideration in the STP (Ref. 1). However, this conclusion did not consider the potential for multiple-consecutive SRV actuations. The design deficiency identified by General Electric resulted from a more detailed transient analysis indicating that several SRVs would experience consecutive actuations following a design basis reactor isolation transient (e.g., closure of all main steamline isolation valves).

In our meeting on October 27, 1977, the Mark I Owners and GE presented the results of a generic assessment of the effects of multiple-consecutive SRV actuations. The Mark I Owners considered the number of valves predicted by the analysis to actuate consecutively to be overly conservative in comparison to operating experience. Therefore, the generic assessment was based on what the Owners' Group considered to be a more realistic estimate of the number of valves which would experience consecutive actuations. The resulting structural response was based on more recent data obtained from the Monticello in-plant SRV discharge tests (Ref. 2).

The staff concluded that the generic assessment did not provide an adequate basis for interim resolution of this issue, since it did not consider the plant SRV configurational differences and there was subjective judgment involved in the application of the Monticello test results. Therefore, on March 20, 1978, letters were sent to each of the Mark I Owners requesting that they perform an interim, plant-specific assessment. These letters specified the criteria to be used to perform the interim assessment and indicated that the structural response should be compared to a limiting strength ratio of 0.5, in accordance with the structural acceptance criteria for the Mark I STP. Continued operation was permitted, based on past operating experience of transient isolation events, during the period while the assessments were being performed.

Table 1 summarizes the licensees' responses and the corrective action taken where it was necessary to satisfy the acceptance criteria for the multiple-consecutive SRV discharge event.

EVALUATION

Each plant-unique structural analysis of the response of the torus and its support system to the loads associated with multiple-consecutive SRV actuations was performed in accordance with the following staff criteria included in the March 20, 1978, letters:

"(1) The number of valves which experience subsequent actuation shall be determined from a plant-unique assessment of the transient which reflects the valves groupings and the SRV setpoints in the facility's Technical Specifications. Variations in the SRV setpoints may be accounted for, provided all of the setpoints are distributed in a manner dictated by actual SRV performance testing. Plants with similar SRV discharge arrangements may be grouped for this assessment, provided their similarity is demonstrated.

"(Although discussions are currently being held between GE and the staff regarding the transient analysis models used to predict the SRV response sequence, we conclude that the current models are acceptable for this interim assessment. The ultimate resolution of this issue in the Long-Term Program will require the use of transient analysis models which resolve staff concerns regarding the current models).

"(2) The plant specific variations to the hydrodynamic characteristics of the SRV discharge line configurations shall be accounted for by the use of a correction factor derived from the SRV discharge analytical model. This factor shall be based on average line conditions for those lines predicted to subsequently actuate, as compared to the Monticello 'Bay D' discharge conditions. The basis for averaging shall be described and justified.

"(3) All available peak structural response data for single SRV discharge events, with approximately the same distances between the discharge point and a point on the structure, should be averaged to obtain the expected values of peak structural response at that point as a function of its distance from the discharging SRV. Certain data may be omitted if it can be demonstrated that such data are inconsistent and should not be considered.

"(4) For structures excited primarily by the overall moments of the torus (e.g., the suction header, the torus support columns, the ring header, etc.), the absolute sum of the structural responses to single SRV actuations shall be used to determine the effects of the same valves actuating simultaneously.

"(5) The consecutive valve actuation factors shall be determined from the Monticello data, or any other available test data, by considering the peak structural responses for an appropriate set of gauges for all consecutive valve actuation tests. For a given set of gauges, the mean plus one standard deviation of all peak structural responses for each gauge shall be computed. These values, in conjunction with the appropriate cold pipe condition structural responses, shall be utilized to compute a set of consecutive actuation factors. These consecutive valve actuation factors shall be averaged to determine one consecutive valve actuation factor which is applicable to the area(s) of the structure for which this set of gauges is appropriate. Certain data may be omitted if it can be demonstrated that such data are inappropriate and should not be considered.

"(6) If the results of this assessment indicate that the limiting strength ratio for either the torus shell or the torus support system is greater than 0.5, corrective measures should be promptly instituted to reduce the limiting strength ratio(s) to less than 0.5. This action may consist of reassigning SRV setpoints, reducing the SRV setpoints, or other measures. If it is determined that corrective measures are necessary for the facility, the submittal should describe proposed corrective measures, including the associated schedule for their completion."

These criteria were developed by the staff from a detailed review of the Monticello test data and with consideration for the uncertainties associated with the models used to predict the number of SRVs which will consecutively actuate. We conclude that these criteria will provide a reasonable estimate of the structural response of the torus and its support system to a multiple-consecutive SRV discharge event. In addition, there are existing conservatisms in the models that are used to predict the number of valves that consecutively actuate and have not been altered.

Because the Monticello test data used for the individual plant assessments is proprietary, GE submitted a generic proprietary report that detailed the methods used to develop plant-specific hydrodynamic and structural correction factors (Ref. 3). This report was subsequently amended (Refs. 4,5) to correct the analysis results and to incorporate comments by the staff pertaining to the SRV discharge line parameters used for certain analyses.

Hydrodynamic multipliers were generated from computer codes developed by GE to predict the peak pressure loads on the torus resulting from SRV discharge. These multipliers were generated to account for SRV discharge configurational differences between all of the individual plants and Monticello. The ratio of the maximum positive pressures was used to adjust the Monticello torus shell response. This factor was corrected for attenuation to produce a multiplier for the Monticello support column response. The staff has concluded that these computer models do not conservatively predict the peak pressure loads. However, because the results of these models are ratioed, we find that the application of these models for this interim assessment is acceptable.

Similar correction factors were derived directly from the Monticello test data to account for structural variations between the facilities. These correction factors were developed and applied in accordance with criteria 3 through 5 above. The resulting SRV-related stresses were combined with the seismic and dead weight loads, and the combined response was then compared to the structural acceptance criteria for the base case analysis in the Mark I STP (Ref. 6).

In a number of cases, plant-specific in-plant test results were used to supplement the generic evaluation technique. The results of relief valve discharge tests performed at Peach Bottom, Pilgrim, Hatch, and Millstone were compared with the results of applying the GE generic evaluation technique. In all cases, the GE multipliers predicted loads on the torus support system that were greater than the loads measured during the tests. However, in two cases, Millstone and Peach Bottom, the tests produced torus shell stresses greater than the stresses predicted from the GE evaluation technique.

The average midbay torus shell at Peach Bottom (measured by strain gauge 2) due to SRV discharge tests of valves "C" and "D" was 80 percent higher than the stress predicted by the GE multipliers. The Peach Bottom facility differs from the Monticello facility in that Peach Bottom has saddle supports and the Peach Bottom relief valve discharges are not located along the centerline of the torus. These effects create higher local bending stresses in the Peach Bottom torus shell than the stresses that occurred in Monticello. The midbay torus shell stress in the Millstone facility (measured by strain gauge R22) for the test of valve "A" was 60 percent higher higher than the stress predicted by the GE multipliers. Examination of the stress components shows that the largest component is in the longitudinal direction, thus indicating a high proportion of bending stress which may not have been considered a primary stress for this evaluation. Nevertheless, we performed an evaluation of the torus shell stresses for those plants using the GE evaluation technique assuming an increase of 80 percent to account for the worst set of measured test data. The results of this evaluation showed that all shell stresses for these plants are within the limiting strength ratio of 0.5.

Prior to the assessment in this report, quencher SRV discharge devices had been installed in the Oyster Creek facility. All other facilities utilize ramshead discharge devices, with the exception of Monticello where quencher devices have recently been installed. In May 1978, the Jersey Central Power and Light Company submitted the results of in-plant tests of the quencher discharge loads for the Oyster Creek facility. Specific tests performed to address multiple-consecutive SRV actuations show that the structural response is within the limits of the ASME Code.

SUMMARY

The results of the plant-specific interim assessments of multiple-consecutive SRV discharge loads on the torus and its support structures show that all plants are within the limiting strength ratio of 0.5 for the combination of weight, seismic, and SRV actuation loads, are in conformance with the acceptance criteria specified for the Mark I Containment Short-Term Program, and are subject to the corrective action specified in Table 1. In addition both Oyster Creek and Nine Mile Point meet the ASME Code limits for column stability and torus shell primary membrane stresses.

In some cases, we did not agree with certain plant-specific assumptions used for the interim assessments. In those cases where the differences were significant, the affected licensees were directed to revise and resubmit their analyses. For the remainder, we performed analyses that showed the differences not to be significant and the limiting strength ratio to be within the acceptance criteria; thus, no action was taken.

In those cases where corrective action was necessary, that action has been completed or will be completed prior to plant startup. Millstone had committed to install column braces by December 1978 to reduce the limiting strength ratio to less than 0.5. The Millstone licensee informed us on December 8, 1978, that this modification has been completed.

The results of these interim assessments demonstrate that each of the operating Mark I BWR facilities, with the necessary corrective action, can accommodate a multiple-consecutive SRV discharge event with sufficient margin to assure the functional performance of the torus and its support structures. On this basis, we conclude that continued operation of these facilities is acceptable until this issue is ultimately resolved as part of the Mark I Long-Term Program.

TABLE 1

MARK I MULTIPLE-CONSECUTIVE SRV INTERIM ASSESSMENT

Plant Name	Submittal Dates	Number of SRVs	Number in MCA*	MCA* Limiting Component	Corrective Action
Browns Ferry 1-3	06-06-78 08-17-78	11	7	Saddles	None
Brunswick 1-2	05-15-78 09-29-78	11	11	N/A	None
Cooper Station	06-13-78 06-29-78	8	8	Column/torus weld	None
Dresden 2-3	06-30-78	5	1	Column base pin	None
Duane Arnold	07-25-78	5	2	Column/torus weld	Stagger SRV setpoints
FitzPatrick	07-07-78 07-31-78 08-18-78 08-25-78 09-28-78 11-14-78	11	2	Column	Stagger SRV setpoints
Hatch 1-2	05-24-78 07-27-78 08-01-78 08-07-78	11	4 & 11	Outer column (Unit 1) Torus shell (Unit 2)	None
Millstone 1	06-05-78 07-31-78 10-04-78 10-23-78 12-08-78	6	6	Column	Strengthen limiting support columns
Monticello	05-24-78	8	8	Torus shell	None
Nine Mile Point	05-26-78 07-26-78	6	6	Column	None

TABLE 1 (continued)

Plant Name	Submittal Dates	Number of SRVs	Number in MCA*	MCA* Limiting Component	Corrective Action
Oyster Creek	06-27-78	5	2	Column	None
Peach Bottom	07-03-78	11	8	Outer column	None
Pilgrim	06-05-78 07-21-78	4	4	Column/torus weld	None
Quad Cities 1-2	06-30-78	5	1	Column/torus weld	None
Vermont Yankee	05-24-78	4	1	Column/torus weld	Stagger SRV setpoints

*Multiple-consecutive actuation (number of valves predicted to consecutively actuate)

REFERENCES

1. U. S. Nuclear Regulatory Commission, "Mark I Containment Short Term Program Safety Evaluation Report," USNRC Report NUREG 0408, December 1977.

2. General Electric Company, "Final Report, In-Plant Safety/Relief Valve Discharge Load Test - Monticello Plant," GE Proprietary Report NEDC-21581-P, August 1977.

3. Proprietary letter from L. J. Sobon, GE, to V. Stello, Jr., NRC, Subject: "Mark I Containment Program Multiple Consecutive S/RV Actuation Evaluation, Task 7.1.3," July 21, 1978.

4. Letter from L. J. Sobon, GE, to V. Stello, Jr., NRC, Subject: "Mark I Containment Program Multiple Consecutive S/RV Actuation Evaluation," August 14, 1978.

5. Letter from L. J. Sobon, GE, to C. I. Grimes, NRC, Subject: "Mark I Containment Program Multiple Consecutive S/RV Actuation Evaluation," November 13, 1978.

6. NUTECH Company, "Description of Short Term Program, Plant Unique Torus Support Systems and Attached Piping Analysis," NUTECH Report MK1-02-012, Revision 2, June 1976.