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

Enclosure 1

JAN 5 1978

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

FROM: L. C. Shao, Chief, Engineering Branch, DOR
W. R. Butler, Chief, Plant Systems Branch, DOR

SUBJECT: EVALUATION OF MULTIPLE-SUBSEQUENT ACTUATIONS OF SAFETY-RELIEF VALVES IN MARK I PLANTS

In letters dated November 1, 1977, each of the Mark I Owners submitted additional information concerning the potential consequences of multiple-subsequent actuations of safety-relief valves (SRVs) during a transient isolation event (i.e., MSIV closure). This information was submitted in response to a staff request for a plant-unique assessment, conveyed to the Owners Group in a meeting on October 27, 1977.

The Plant Systems Branch and the Engineering Branch have reviewed these submittals in conjunction with the information presented by General Electric (GE) in the October 27, 1977 meeting and in subsequent telephone conversations. As a result of this review, we have made the following observations regarding both the transient analyses used to predict the number of valves which will subsequently actuate and the extrapolation of load measurements from the Monticello in-plant SRV tests:

1. GE performed sensitivity analyses with the SAFE code to determine the variation in the number of valves which are predicted to subsequently actuate to changes in MSIV closure time, power reduction, SRV opening delay and opening stroke time, and SRV opening setpoint. Both the SRV opening characteristics and the SRV setpoint were found to have a significant effect on the prediction of the number of valves which experience a "second pop".

2. The plant unique variations due to differences in SRV line size, submergence, vacuum breaker size, etc. were determined by the application of GE's current analytical model for SRV discharge loads. From this model, plant-unique multipliers were developed which were normalized to the Monticello test conditions. These multipliers range from 0.3 to 1.39. However,

this model has not been accepted by the Containment Systems Branch since it has not been demonstrated that the loads predicted by the model are conservative. Further, the model does not have the capability to predict the loads resulting from a second actuation.

3. In the Monticello test report, there are significant variations in the structural responses at a given point, from test to test, for similiar test conditions.

4. The data base resulting from the Monticello tests is not sufficient to perform a statistical evaluation to determine the probability distributions for either (1) the structural responses for similiar test conditions, or (2) the manner by which structural responses for single SRV actuations combine when compared to several SRVs discharging simultaneously.

5. In assessing the effects of multiple actuations, the structural responses to single SRV actuations do not combine consistently at various points on the structure, when compared to the same valves discharging simultaneously; the structural responses at various points on the structure vary from less than SRSS (i.e., square root of the sum of the squares) to greater than the absolute sum of the responses for the same valves discharging individually.

6. GE's "most probable" estimate of the number of valves which experience a second pop is three. However, in more than one hundred actual isolation transients, only two events occurred where more than one valve experienced a second actuation. GE made this observation and also indicated that in some cases the records were difficult to interpret. In none of these cases was there any evidence of structural damage of the containment shell.

Based on the foregoing observations, we have prepared the enclosed staff position regarding the short-term assessment of multiple-subsequent SRV actuations. Although there are a number of uncertainties involved in these calculations, which cannot be resolved within the time frame necessary to complete this assessment, we

believe that the enclosed position will provide a reasonable estimate of the effects of multiple-subsequent SRV discharges following a transient event. There are some inherent conservatisms in this approach which are not quantifiable, but which appear to have been demonstrated by the plant operating experience.

We recommend that the enclosed position be transmitted to each Mark I owner for action.

We have discussed the results of our review with representatives of the Containment Systems Branch, DSS, and they are in agreement with the enclosed staff position.

W R Butler

W. R. Butler, Chief
Plant Systems Branch
Division of Operating Reactors

L. C. Shao

L. C. Shao, Chief
Engineering Branch
Division of Operating Reactors

Enclosure:
As stated

Contact:
K. Herring, EB/DOR
x28066
C. Grimes, PSB/DOR
x28077

cc: V. Stello
R. Mattson
K. Goller
R. Tedesco
J. Knight
G. Lainas
J. Kudrick
N. Su
C. Anderson
J. Guibert
R. Stuart
K. Herring
C. Grimes
I. Sihweil
L. Shao
W. Butler
B. Buckley

STAFF POSITION ON THE ASSESSMENT
OF MULTIPLE - SUBSEQUENT SRV
ACTUATIONS

We have reviewed your submittal dated November 1, 1977 regarding the potential consequences of multiple-subsequent SRV actuations following a primary system isolation transient. Based on our review, we have found that, in general, the techniques used for this assessment were inappropriate. We have, therefore, developed the following criteria for use in plant-unique assessments of this concern as it applies to Mark I BWR facilities. You are requested to submit a plant-unique assessment of this concern for your facility within 60 days of receipt of this letter.

Your submittal should include a description of the methods used to satisfy the following criteria. Where appropriate, plant unique data may be used for this assessment, provided that the test procedures and data are documented.

(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 your 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.

(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) The effects of a multiple valve discharge event, as determined from the data on individual SRV discharges, shall be determined by taking the SRSS of the individual valve effects and increasing this value by 20 percent, except as noted in (5) below.

(5) For structures excited primarily by the overall movements 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.

(6) 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 gages for all consecutive valve actuation tests. For a given set of gages, the mean plus one standard deviation of all peak structural responses for each gage 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 acuation factor which is applicable to the area(s) of the structure for which this set of gages is appropriate. Certain data may be ommitted if it can be demonstrated that such data are inappropriate and should not be considered.

(7) 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 then 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 you determine that corrective measures are necessary, for your facility, your submittal should describe proposed corrective measures, including the associated schedule for their completion.

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

Enclosure 2

MEMORANDUM FOR: A. Schwencer, Chief, ORB#1
D. Davis, Acting Chief, ORB#2
G. Lear, Chief, ORB#3
R. Reid, Chief, ORB#4
D. B. Vassallo, Assistant Director for Light Water Reactors
Division of Project Management

FROM: K. R. Goller, Assistant Director for Operating Reactors
Division of Operating Reactors

SUBJECT: ASSESSMENT OF MULTIPLE-SUBSEQUENT ACTUATIONS OF SAFETY-RELIEF VALVES IN MARK I PLANTS

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 product line. In a letter dated October 11, 1977 (G. Sherwood, GE to N. Mosley, NRC) and during a meeting between members of the Mark I Owners Group, GE, and the NRC staff on October 27, 1977, the implications of this design deficiency to operating BWR facilities were discussed. As a result, the staff requested that each utility submit a basis for continued plant operations by November 1, 1977.

Our review of the material submitted to date has indicated the need for a plant-unique assessment of the effects of multiple-subsequent SRV actuations. We have prepared the enclosed sample letter for transmittal to licensees of operating BWR Mark I facilities requesting such an assessment and providing criteria for the conduct of the assessment.

Assigned project managers for operating Mark I facilities should transmit this letter to the licensees by March 10, 1978. The sample letter is available on the Vydec machine.

We recommend that this letter also be transmitted to the applicant for Hatch Unit No. 2.

This position has been closely coordinated with and concurred in by the AD/PS and AD/RS of DSS and their staffs.

Should you have any questions relating to this action, contact J. Guibert (x28256) or Chris Grimes (x28077).

K. R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Enclosure:
Sample letter to licensees

cc: See Page 2

cc w/enclosure:
V. Stello
D. Eisenhut
K. Goller
W. Butler
L. Shao
R. Tedesco
G. Lainas
B. Buckley
R. Stuart
J. Kudrick
J. Guibert
T. Wambach
M. Fletcher
R. Bevan
R. Snaider
P. O'Connor
C. Trammell
R. Clark
S. Nowicki
J. Hannon
D. Verrelli
G. Vissing
M. Fairtile
C. Thomas
C. Grimes

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

Docket No.

(Utility)

Gentlemen:

RE: MULTIPLE-SUBSEQUENT ACTUATIONS OF SAFETY/RELIEF VALVES FOLLOWING AN ISOLATION EVENT

In a meeting on October 27, 1977, the General Electric Company (GE) and the Mark I Owners Group provided the staff with the results of an assessment of the effects of multiple-subsequent actuations of safety/relief valves (SRVs) following an isolation event. This assessment was provided to justify the deferral of this issue until its ultimate resolution as a part of the Mark I Containment Long-Term Program. At the conclusion of that meeting, the staff requested that each utility submit a basis for continued operation by November 1, 1977 including a description of any interim corrective measures which may be implemented. The staff further indicated that it may require plant-unique assessments to be provided in the near future. A number of the submittals made on November 1, 1977 contained additional information relative to the effects of multiple-subsequent SRV actuations.

The assessments that we have received to date have been based on an application of the results of the Monticello SRV discharge (ramshead) tests. During the course of our review of the Monticello test results, we have noted that there are significant variations in the measured structural responses for similar test conditions. As a result, we have concluded that the data base is insufficient to determine the probability distribution for either (1) the structural responses for similar test conditions, or (2) the manner by which structural responses for single SRV actuations are to be combined in determining the structural response to several SRVs discharging simultaneously. Further, in assessing the effects of multiple SRV actuations, the structural responses to single SRV actuations do not combine consistently at various points on the structure, when compared to the responses for the same valves discharging simultaneously.

We recognize that, at the present time, the Monticello test results provide the best available data for determining the effects of multiple-subsequent SRV actuations. However, the application of the Monticello test results involves a considerable amount of subjective judgment. We have, therefore, developed the enclosed criteria, based on our interpretation of the Monticello data, which we believe will provide a "most probable" estimate of the effects of an isolation transient event. In our view, such an estimate is consistent with the philosophy of the Mark I Containment Short-Term Program and is acceptable on an interim basis, while the Long-Term Program is being conducted.

The enclosed criteria should be used to perform a plant-unique assessment of this concern as it relates to Mark I BWR facilities. You are requested to submit this assessment for your facility within 60 days of the receipt of this letter. Since over 100 of these transient events have occurred for which only two events resulted in multiple-subsequent SRV actuations, and since no evidence of structural deterioration was found, we conclude that continued operation is acceptable while this assessment is being performed. Your submittal should include a description of the methods used to satisfy these criteria. Where appropriate, plant-unique data may be used for this assessment, provided that the test procedures and data are documented.

Sincerely,

Branch Chief

Enclosure:
Criteria for the Assessment
of Multiple-Subsequent SRV
Actuations

Enclosure

CRITERIA FOR THE ASSESSMENT
OF MULTIPLE-SUBSEQUENT SRV ACTUATIONS

1. The number of valves which experience subsequent actuation shall be determined from a plant-unique assessment of the transient which reflects the valve groupings and the SRV setpoints in your 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. The effects of a multiple valve discharge event, as determined from the data on individual SRV discharges, shall be determined by taking the SRSS of the individual valve effects and increasing this value by 20 percent, except as noted in (5) below.

5. For structures excited primarily by the overall movements 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.

6. 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.

7. 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 you determine that corrective measures are necessary, for your facility, your submittal should describe proposed corrective measures, including the associated schedule for their completion.