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

JUL 9 1980

Mr. Milton S. Plesset, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Plesset:

Subject: Response to Inquiry Concerning BWR Relief Valve Discharge Piping

In your letter to the Commission, dated February 13, 1980, you requested that the NRC staff look into specific matters relating to safety-relief valve (SRV) discharge piping in operating BWR facilities with the Mark I containment design. The purpose of this letter is to respond to that request and related issues raised during the Committee's 238th meeting.

The Committee requested that the staff assure the adequacy of the requirements for verification of the design, fabrication, and in-service inspection of the Mark I containment modifications and, in particular, the SRV discharge piping in the suppression chamber airspace. The Mark I containment modifications will be described in plant-unique analyses for each facility. These analyses will demonstrate conformance with the staff's acceptance criteria, which the Committee endorsed in the February 13, 1980 letter. The design of structures will be compared to the allowable stresses in Section III, Division 1, Subsection NE of the ASME Boiler and Pressure Vessel Code for steel structures, and Section III, Division 2, of the ASME Boiler and Pressure Vessel Code for steel lined concrete structures. The design of Class 2 and 3 piping systems will be compared to Section III, Division 1, Subsections NC and ND of the ASME Boiler and Pressure Vessel Code. The staff will review the completed plant-unique analyses to assure the adequacy of the modification designs and to assure that the margins of safety for the overall containment design have been restored.

Requirements for fabrication and in-service inspection of containment components are currently being developed by the ASME Code Containment Working Groups. The staff is represented on these working groups. In the interim, the staff has provided assistance to the Mark I Owners Structural Working Group in the formulation of general guidance for the Mark I-related modifications. We are confident that this general guidance, in conjunction with the routine quality control inspections by IE, will provide adequate verification of the fabrication and inspection of the Mark I containment modifications. Specific requirements for in-service inspection applicable to the SRV discharge piping are not presently addressed in the ASME Code. However, the Section XI working group is currently developing in-service inspection requirements for the containment. When these requirements have been approved, we will incorporate the appropriate portions of the SRV discharge piping into the in-service inspection programs. In the

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interim, periodic visual inspections are being conducted to assure the continued integrity of the SRV discharge piping.

The Committee also requested that the staff investigate the potential for and the consequences of a failure of the SRV discharge piping in the suppression chamber airspace for the existing designs. We, subsequently, transmitted a request to the Mark I Owners Group (Enclosure 1) to obtain the information necessary to perform such an assessment. General Electric submitted a generic response (Enclosure 2) on behalf of the Mark I Owners Group. To determine the potential for such failures, General Electric screened out fourteen of the twenty-two operating Mark I plants which have already established that the SRV discharge piping conforms to the acceptance criteria for the Long Term Program. For the remaining eight plants, General Electric estimates that the minimum margin to failure by plastic collapse is approximately a factor of three. We have reviewed the bases for this assessment and the related test data, and we conclude that it reflects a reasonable estimate of the minimum margin to failure. Further, we consider this margin adequate during the interim period while the Mark I modifications are being performed.

With regard to the consequences of an SRV discharge line failure in the suppression chamber airspace, General Electric has stated that they consider this event to be outside the scope of the design basis, because it involves both an active and a passive failure subsequent to an initiating event.

Although we presently do not have the manpower available to perform a comprehensive analysis of such an event, we have performed scoping analyses. These analyses suggest that should a break occur in the suppression chamber airspace in conjunction with a stuck-open SRV, the resultant containment pressure would probably exceed the design value; however, with consideration for the heat absorption capability of the containment structure, internal structures, suppression pool surface, and containment sprays, the resultant pressure would probably be within the ultimate strength of the containment structure (approximately twice design pressure). Manually discharging through other SRV's and similar rapid cool-down techniques can further mitigate the consequences of such an event.

Based on the results of the screening analyses described by General Electric and our consequence evaluation, we conclude that no further action is warranted on this matter at this time. We will continue to keep the Committee informed as new information becomes available.

Sincerely,

Original Signed by
H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Letter from D. G. Eisenhut, NRC, to R. H. Logue, Mark I Owners Group dated 3/18/80
2. Letter from R. H. Buchholz, GE, to D. G. Eisenhut, NRC dated 5/2/80

ENCLOSURE 1

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

MAR 18 1980

Mr. R. H. Logue, Chairman
Mark I Owners Group
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Dear Mr. Logue:

In the attached letter from the Advi y Committee on Reactor Safeguards, dated February 13, 1980, the ACRS h requested that the NRC staff investigate the potential for and consequences of a failure in the safety-relief valve (SRV) discharge lines located in the suppression chamber airspace. This evaluation is to be performed for the existing discharge line configurations in the interim period while the Mark I Long Term Program modifications are being performed.

To perform this evaluation would require that we send requests for detailed design information and analyses to each Mark I licensee. However, to simplify and expedite the resolution of this issue, we request that the Owners Group develop a generic response on behalf of all of the Mark I licensees. The generic assessment should be performed in the following manner:

1. Screen the discharge line configurations in the suppression chamber airspace (between the connection at the entry to the suppression chamber and the discharge device support) and estimate the minimum margins to failure for the design-basis discharge reaction loads. In the event that particular discharge line configurations are found to be atypically weak, the individual licensees should respond directly.

2. Provide a best-estimate of the containment response for a broken SRV discharge line in the suppression chamber airspace with a stuck-open SRV on the broken line. This analysis should consider condensation and heat transfer to the containment structures and the suppression pool surface.

The results of this interim assessment should be submitted within 45 days of your receipt of this letter. The response should include a brief description of the discharge line configurations and the assumptions used in the containment response analysis. Should you have any questions concerning this request, contact C. Grimes (301-492-8204).

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

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ENCLOSURE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

February 13, 1980

Honorable John F. Ahearne
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: NRC ACCEPTANCE CRITERIA FOR THE MARK I CONTAINMENT LONG TERM PROGRAM

Dear Dr. Ahearne:

During its 238th meeting, February 7-9, 1980, the Advisory Committee on Reactor Safeguards reviewed the NRC Acceptance Criteria for the Mark I Containment Long Term Program. This matter was considered at ACRS Fluid Dynamics Subcommittee meetings held on May 23, 1978, November 28-30, 1978, September 13-14, 1979, and November 16, 1979. During its review, the Committee had the benefit of discussions with representatives of the NRC Staff and the Mark I Owners Group.

The NRC Acceptance Criteria for the Mark I Containment Long Term Program are intended to establish design basis loads that are appropriate for the anticipated life of each Mark I BWR facility and to restore the originally intended design safety margins to each Mark I containment system.

The Mark I program was initiated in 1975 in response to loss of coolant accident and safety relief valve (SRV) dynamic loads identified by the General Electric Company during the course of performing large scale testing for the Mark III pressure-suppression containment in 1972-1974. A period of reevaluation resulted in issuance of the Short Term Program Acceptance Criteria in December 1975 which established interim design bases for continued operation of the Mark I BWRs. The Acceptance Criteria for the Long Term Program have been developed from a program of small and full scale tests in two and three dimensional geometries.

The Mark I Owners submitted proposed loads in the "Mark I Containment Program Load Definition Report" in December 1978 and detailed the methods to be used in plant unique analyses in the "Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Applications Guide." Following review of the available information, the NRC Staff determined that certain changes and clarifications to the criteria proposed by the Mark I Owners were necessary. The NRC Staff technical requirements were delineated in the "NRC Acceptance Criteria for the Mark I Containment Long Term Program" issued in October 1979 and also in several additions to the acceptance criteria as discussed during the 238th ACRS meeting. The additions to the Acceptance Criteria were intended, in part, to alleviate some of the difficulties the Mark I Owners had in calculating credible structural responses to SRV actuations.

The Committee recognizes the thoroughness of the efforts taken by the NRC Staff and the Mark I Owners to resolve the generic Mark I issues and believes that the NRC Acceptance Criteria and additions, as proposed, provide a suitably conservative basis for performing the Long Term Mark I Containment structural response analyses. The Mark I Owners indicated that they continue to have significant difficulty in calculating credible structural responses to some SRV loads and they would like to continue to work with the Staff on a generic basis to resolve these difficulties. The NRC Staff would like to complete the generic Mark I program and resolve any remaining problems as they arise from the plant unique analyses. The Committee believes that the individual Mark I Owners can work with the Staff to resolve any additional difficulties that may arise from the plant unique analyses as modifications are being made to the containment structures.

The Committee believes that the Staff should assure the adequacy of the requirements for verification of the design, fabrication, and inservice inspection of the Mark I containment modifications and, in particular, the SRV discharge piping in the wetwell airspace. Further, in the interim period while the Mark I modifications are being performed, the Staff should investigate the potential for and consequences of a failure in the SRV discharge piping in the wetwell airspace for the existing designs. The Committee wishes to be kept informed on this matter.

The Committee believes that, with due consideration to the above items, the generic Mark I Long Term Program can be concluded and the modifications to the individual Mark I BWRs can be implemented on a reasonable schedule over the next 18 months.

Sincerely,

Milton S. Plesset

Milton S. Plesset
Chairman

References:

1. General Electric Company, "Mark I Containment Program Load Definition Report," Revision 0, NEDO 21888, December 1978.
2. General Electric Company, "Mark I Containment Program Structural Acceptance Criteria Plant Unique Analysis Applications Guide," NEDO 24583, December 1978.
3. U.S. Nuclear Regulatory Commission, "NRC Acceptance Criteria for the Mark I Containment Long Term Program," October 1979, and additions included in the February 8, 1980, transcript of the 238th ACRS Meeting.

ENCLOSURE 2

. NERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
Mail Code 682, Telephone (408) 925-5722

RHB 033-80 MFN 091-80

May 2, 1980

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Mr. D. G. Eisenhut, Director
Division of Operating Reactors

Gentlemen:

SUBJECT: MARK I CONTAINMENT PROGRAM -
POSTULATED FAILURE OF A SAFETY/RELIEF VALVE
DISCHARGE LINE

Reference: Letter, D. G. Eisenhut (NRC) to R. H. Logue
(Philadelphia Electric Co.), dated March 18, 1980

The reference letter contained an NRC request that the Mark I Owners Group investigate a postulated failure in the safety/relief valve (S/RV) discharge line located in the suppression chamber airspace. General Electric Company is providing this letter on behalf of the Mark I Owners Group as part of the Mark I Containment Program. This correspondence is the generic response requested by the NRC in the reference letter.

The design basis for Mark I plants consists of an initiating event (e.g. a loss-of-coolant-accident) coincident with an assumed failure of a single active component. An event such as the one postulated in the reference letter assumes that an initiating event has occurred (e.g. reactor vessel pressurization due to an isolation transient) followed by an opening of an S/RV, a subsequent active failure (S/RV does not close) and an additional passive failure (safety/relief valve discharge line break). This is beyond the design basis of the plant. Therefore, analysis of the consequences of this event is not warranted and is not presented. The attachment to this letter provides additional justification that the postulated scenario need not be considered. Included is a discussion of the very low probability of such an event and an evaluation which confirms the mechanical integrity of the safety/relief valve discharge line (S/RVDL) piping in the wetwell airspace as currently configured.

The attachment to this letter demonstrates that typical Mark I plant S/RVDL piping can easily tolerate anticipated operating conditions and no failures are expected. Furthermore, the evaluations and plant modifications resulting from application of the Mark I Program Structural Acceptance Criteria (General Electric Report, NEDO-24583-1, "Mark I Containment Program, Structural Acceptance Criteria, Plant Unique Analysis Application Guide" dated October 1979) provide confidence that increased plant margins will be implemented as part of the Mark I Long Term Program.

The information presented in this correspondence shows that present plant S/RV discharge line configurations have adequate margins in the interim period until utility implementation of the Mark I Long Term Program objectives is complete.

Very truly yours,

S. J. Stark for R. H. Buchholz

R. H. Buchholz, Manager
BWR Systems Licensing

Enclosure

cc: [redacted]
L. S. Gifford (GE-Bethesda)

ATTACHMENT

GENERIC EVALUATION OF MARK I SAFETY/RELIEF VALVE DISCHARGE LINE INTEGRITY

This attachment presents a generic evaluation of Mark I Safety/Relief Valve Discharge Line (S/RVDL) integrity. Included is a discussion of the probability of a failure of a Mark I S/RVDL in the wetwell airspace concurrent with a stuck-open S/RV in the same line. Also presented is an evaluation which confirms the mechanical integrity of Mark I S/RVDL piping in the wetwell airspace as currently configured.

I. EVENT PROBABILITY

The postulated failure of a Mark I S/RVDL in the wetwell airspace concurrent with a stuck-open S/RV in the same line has a very low probability of occurrence. There are many factors affecting the probability of such an event. For example, the S/RVDL pipe failure of concern in the postulated event is credible only if the pipe is pressurized. But an S/RVDL is pressurized for only a short period of time following S/RV actuation. Peak S/RVDL pressure occurs only for a fraction of a second while the water leg is being cleared from the line. Pipe pressures are much lower during steady state discharge. The fact that only the fraction of the S/RVDL piping in the wetwell airspace is of concern is also considered. Likewise, the possibility of a stuck-open S/RV exists only while a plant is operating, therefore, a typical plant availability factor is included. Using a typical pipe failure rate from WASH-1400 and additional factors such as those discussed above results in a calculated event probability of less than 10^{-7} per plant per year. Thus, analysis of the consequences of such an event is not warranted.

II. S/RVDL MECHANICAL INTEGRITY

In screening the Mark I plants for determining representative S/RVDL configurations for evaluation, those plants which have completed S/RV modifications and/or analyses for this evaluation basis in accordance with the Plant Unique Analysis Application Guide (PUAAG), or which do not have their operating license need not address this issue. Compliance with the criteria in the PUAAG makes the need to perform an "interim" evaluation unnecessary. The plants included in this category are:

Browns Ferry 1, 2 and 3

Millstone

Fermi 2

Hope Creek 1 and 2

Cooper

Dresden 3

FitzPatrick

Oyster Creek

Nine Mile Point

Quad Cities 2

Hatch 1 and 2

Peach Bottom 2 and 3

Of the remaining plants, two groupings were established: Group 1 consists of those plants which have essentially straight pipe in the portion of the S/RVDL located in the wetwell airspace. Group 1 consists of the following plants:

Monticello

Brunswick 1 and 2

Duane Arnold

Pilgrim

Quad Cities 1

Vermont Yankee

Group 2 consists of those plants which have one or more elbows in the wetwell airspace. Group 2 consists of the following plant:

Dresden 2

ANALYSIS OF GROUP I PLANTS

In order to assess the S/RVDL performance characteristics during an S/RV discharge for Group 1 plants, use was made of observations from the Monticello in-plant S/RV tests as follows:

S/RVDL PIPE STRESS

Monticello Ramshead test data was used to assess the S/RVDL characteristics during S/RV discharge. The pipe stresses from the Ramshead tests are representative of Group 1 Ramshead installations. The highest measured stresses for the S/RV piping in the wetwell for normal operating loads was 16.3 ksi. Subsequent to the Ramshead test, T-Quenchers were installed along with additional supporting hardware such that measured pipe strains were reduced more than 50 percent.

The test value of 16.3 ksi was increased to account for the fact that the test was performed at a reactor pressure of 985 psi which is less than the S/RV setpoint of 1130 psi. To compute piping stresses for S/RV actuation at the setpoint of 1130 psi, the stresses in the line were increased by 8.7%. Sensitivity studies show that this would be the increase in the maximum thrust force for a 145 psi increase in reactor pressure. The maximum measured stress to be used for evaluating the straight pipe is therefore 17.7 ksi.

S/RVDL PIPE PRESSURE

An S/RVDL pipe pressure value of 285 psia was used for this analysis as derived from the Monticello Ramshead test data. This value is considered to be adequately conservative because the corresponding maximum pipe pressure near the water surface for the T-Quencher test was 240 psia.

Minimum margins to failure were determined for a typical straight S/RV discharge line in the wetwell airspace using the equations for analysis provided in Subsection ND of the ASME Boiler and Pressure Vessel Code, but considering actual material properties rather than Code design basis values to estimate failure loads. Yield and ultimate tensile strength properties for the material commonly used for the discharge lines (A-106, Gr. B) were obtained from available material certification data.

The minimum margin for gross ductile rupture was evaluated considering the effects from internal pressure and applied mechanical loads. This margin to failure was computed as the ratio of the calculated loading as described above to the plastic collapse load for the section based on a flow stress equal to the average of the yield and ultimate tensile strengths. The resulting strength ratio was determined to be 0.26. The load source which could cause gross rupture of the piping is the internal pressure load. It should be noted that the stresses from this load when applied to a 10 inch schedule 80 pipe would be 1.2 ksi. Thus, the major portion of the loading is caused by the reaction to the applied mechanical loads.

Additional evaluations were performed for Vermont Yankee since it has a 45^{o} elbow near the pool water surface whereas all other Group 1 plants have the elbow located beneath the pool surface. For this evaluation, Monticello test data provides a stress at the elbow of 12.9 ksi which was increased to 14.0 ksi for operating conditions. The mechanical load associated with the 14.0 ksi stress level when combined with pressure, results in a strength ratio of 0.29 for the elbow.

ANALYSIS FOR GROUP 2 PLANTS

The approach utilized to evaluate the Group 2 plant was similar to that employed for the Group 1 evaluation. However, the applied mechanical loads were established based on analysis rather than test results. The resulting strength ratios are less than the Short Term Program criterion of 0.5. The most limiting strength ratio for this group is 0.30.

III. CONCLUSION

The above informatic monstrates that typical Mark I plant S/RVDL piping in the wetwell airspace can easily tolerate anticipated operating conditions and no failures are expected. The evaluations and plant modifications resulting from app.ication of the PUAAG provide confidence that increased plant margins will be implemented as part of the Mark I Long Term Program. Therefore, present plant S/RV discharge line configurations have adequate margins in the interim period until utility implementation of the Mark I Long Term Program objectives is complete.