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

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MEMORANDUM FOR: T. Novak, Chief, Reactor Systems Branch, DSS

FROM: G. Lainas, Chief, Containment Systems Branch, DSS

SUBJECT: DRAFT SAFETY EVALUATION FOR ANTICIPATED TRANSIENTS WITHOUT SCRAM CONTAINMENT SYSTEMS

The Containment Systems Branch has prepared the enclosed draft Safety Evaluation for the ATWS containment-related aspects for LWR's and includes a bibliography of the vendor's reports that are used in our evaluation. Our evaluation is restricted to Class B plants as defined in WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooler Power Reactors."

As a result of our evaluation, we find that the PWRs meet the acceptance criteria for containments as stated in WASH-1270; i.e., the containment design temperature and pressure are not exceeded. We cannot conclude on the adequacy of GE's proposed limits for the suppression pool temperature for these plants using the rams head device. We can conclude that the temperature limits for these plants utilizing the quencher design (all Mark IIIs) are acceptable. GE has been advised of our concerns and will be responding.

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3.2.4 Acceptance Criteria

3.2.4.3 Containment

The licensing position contained in WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," requires, for those plants defined as Class B, that "calculated maximum containment pressure should not exceed the design pressure of the containment structure. Equipment located within the containment that is relied upon to mitigate the consequences of ATWS should be qualified by testing in the combined pressure, temperature, and humidity environment conservatively predicted to occur during the course of the event."

A further requirement has been identified for BWR pressure suppression containments. Reactor operating experience has indicated that potential instabilities in quenching of relief valve discharge flow could occur at certain steam mass flux conditions and suppression pool temperatures. Therefore, this region of mass flux/temperature where instabilities might occur must be avoided during the ATWS event.

Part B

Babcock & Wilcox

Staff Evaluation and Conclusions

The December 9, 1975 status report provided our evaluation of the B&W ATWS containment transients described in BAW-10099(1). The most severe containment transient was identified as the stuck open safety relief valve ATWS (2) resulting in a peak containment pressure of 10.6 psig at 1300 seconds. Since then the NRC has required that certain single failures also be included in the ATWS evaluation. We have performed an evaluation of the effect which these postulated single failures would have on containment transients using the information contained in reference (1) and (2).

B&W has analyzed the effect of increasing the safety relief valve mass and energy release by a factor of two in BAW-10099(1) to show the sensitivity of additional mass and energy releases to the containment. This transient results in a peak pressure of approximately 25 psig and an integrated energy release of approximately 507 x 10^6 BTU at 1600 seconds. Although this energy release exceeds the integrated LOCA release (typically 450 x 10^6 BTU at 150 seconds and a peak pressure of 45 psig) the rate of release is substantially less.

The event (i.e., ATWS combined with a single failure assumption) that results in the highest energy release to the containment is the postulated ATWS of a loss-of-offsite power with the single failure of a relief valve to close. To determine the containment transient for this event, we added the energy release from a loss-of-offsite power (contained in reference 1 and 2) to the energy release due to a stuck open relief valve (contained in reference 1 and 2). The results show a lower energy release for this event than for the LOCA or doubled flow of a stuck open pressurizer safety relief valve, both of which result in containment transients which are within the containment design parameters. On this basis we conclude that the typical Babcock & Wilcox plant meets the acceptance criteria for containment response as stated in WASH-1270.

Part C

Combustion Engineering

Staff Evaluation and Conclusions

In our status report of December 9, 1975, we provided our review of the ATWS events described in CENPD-158(1) which would result in containment transients. The most severe ATWS transient results showed only a small fraction of the Reactor Coolant System inventory being released to the containment. However, the matter of identifying the most severe containment transient with the inclusion of certain single failures was not resolved. Since then, Combustion Engineering has provided additional information in Revision 1 to CENPD-158 (2).

The complete loss of feedwater ATWS with a stuck open safety relief valve has been identified as resulting in the most severe containment transient. The peak calculated pressure was 24.8 psig which is more than 50% less than the typical plant design pressure of 50 psig. In calculating this peak pressure, no credit was taken for heat removal by containment fan coolers nor passive heat sinks. The integrated energy released to the containment was approximately 134×10^6 BTU in comparison with a typical LOCA integrated energy release of approximately 450×10^6 BTU.

We have considered the conservatisms incorporated in the Combustion Engineering containment response calculations and we have compared the significantly lower energy release of the ATWS events with that of a

LOCA. On this basis, we conclude that the typical Combustion Engineering plant meets the acceptance criteria for containment response as stated in WASH-1270.

Part D

In the December 9, 1975, status report, we provided our evaluation of the GE analysis of the most severe ATWS event, i.e., closure of all mainsteam isolation valves (MSIV). This event also results in a calculated local suppression pool temperature of 173°F. We believed at that time that the local pool temperature should not exceed 160°F until GE could provide additional information to justify a higher pool temperature limit.

For the containment pressure response to ATWS events, the GE analyses show that the calculated containment pressure rises were from 5 to 15 psi for Mark I and II containment and from about 2.5 to 5 psi for Mark III containment. The containment pressures correspond approximately to the increase in vapor pressure due to the temperature rise of the suppression pool. For all ATWS events, the containment pressure is below the containment design pressure by a substantial margin.

GE submitted a topical report NEDE-21078 entitled "Test Results Employed by GE for BWR Containment and Vertical Vent Load."[1] The intent of this report was to provide test results to justify (1) the design of a quencher device, (2) the suppression pool temperature limits and

(3) the vertical downcomer lateral loads used in the design of Mark I containments. With regard to pool temperature limits, this report describes the results of tests conducted in a foreign country to investigate the effect of pool water temperature on steam condensation. Various configurations of steam line discharge devices including a straight pipe and several quencher devices were tested. Our evaluation of the suppression pool temperature limits is based on the information provided in this report.

The pressure suppression pool acts as a heat sink to condense steam discharged from the relief valve during the ATWS events. Temperature limits are required because at high temperatures steam quenching instabilities occur and can induce severe localized pressure loads on structures. The purpose of the tests was to determine the temperature at which the onset of these instabilities occur. Local pool temperatures as used in this evaluation are those temperatures that were measured in the test facility and which we consider to be within 30 feet of the device in the actual plant. Since the temperature distribution in the pool cannot be calculated; only average pool temperatures, a 10°F difference between average and local temperature was considered to relate test conditions to calculated actual plant conditions. This temperature difference has been noted in actual plant operation. The results of our evaluation of the temperature limits for quencher devices and ramshead devices are presented as follows.

Quencher Device

The quencher device consists of four arms about 5 feet long with holes to distribute air and/or steam evenly in the suppression pool. Currently, the quencher device is included in the design of all Mark III type containments and some Mark II containments.

Based on these test results, GE proposed in the "General Electric ATWS Report" dated June 30, 1976,[2] that the pool temperature limit for plants utilizing a quencher device is 200°F local pool temperature or equivalent to 190°F calculated average pool temperature. As a result of our evaluation of the test results presented in NEDE-21078, we find that the test data are sufficient to support the pool temperature limit for the quencher. Therefore, we conclude that the 190°F calculated average pool temperature is acceptable for the containments utilizing the quencher device. It should be noted that the Mark III containment design temperature is 185°F. Therefore, for the Mark III a design temperature of 185°F should be used.

Ramshead Device

The ramshead design consists of a tee or two 90°F - elbows. Currently, ramshead designs are used in most of the Mark I and II type containments.

No specific tests for ramshead design has been performed to establish the pool temperature limits to prevent steam condensation instabilities. GE, however, has used the test results obtained from straight pipes to establish the temperature limit for the ramshead. The topical report NEDE-21078 provides a summary of the straight pipe test results. Based on these test results, GE proposed in the "General Electric ATWS Report" dated June 30, 1976,[2] that the pool temperature limit for plants utilizing a ramshead is 170°F local pool temperature or equivalent to 160°F calculated average pool temperature. We have reviewed this report and had discussions with GE and conclude that the information is not complete enough for us to agree with GE's proposed limits. GE will provide us with additional justification and we will complete our review at that time.

Part E

Westinghouse

Staff Evaluation and Conclusions

Our evaluation of containment related affects of ATWS events described in the Westinghouse topical report WCAP-8330(1) was provided in the December 9, 1975 status report. The results of the analyses performed by Westinghouse (2) showed that a maximum containment pressure of 13.7 psig would occur for the inadvertant depressurization ATWS. In the status report, we concluded that the containment design pressure ($\geqslant$ 50 psig) of a typical Westinghouse plant would not be exceeded by an ATWS event.

At the NRC's request, Westinghouse has since included consideration of a single failure in the ATWS analyses (3). The complete loss of feedwater transient with a stuck open safety relief valve has been identified as resulting in the most severe containment transient. Within the first 250 seconds approximately 90×10^6 BTU are released to the containment. The original analysis (loss-of-feedwater) resulted in a 60×10^6 BTU release at this point and was compared with a LOCA energy release of approximately 450×10^6 BTU. At this point in time, the energy release rate is still 10^5 BTU/sec. For a LOCA, the energy release rate is three or four times higher. However, containment systems are designed to handle these energy release rates and reduce the containment pressure.

We have considered the comparison of ATWS and LOCA energy release as well as previous calculations which show that a LOCA results in the more severe containment transient. On this basis, we conclude that the typical Westinghouse plant evaluated meets the acceptance criteria for containment response as stated in WASH-1270.

REFERENCES

B&W

1. BAW 10099, "Babcock & Wilcox Anticipated Transients Without Scram Analysis," December 1974.

2. Letter, James F. Malley to Victor Stello, Jr., dated January 2, 1975.

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1. CENPD-158, "Analyses of Anticipated Transients Without Scram in Combustion Engineering NSSS's," September 1974.

2. CENPD-158, Revision 1, May 1976.

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1. WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analyses," August 1974.

2. Letter, C. Eicheldinger to D. B. Vassallo, dated May 1, 1975.

3. Response to NRC request for additional ATWS analysis.

GE

1. NEDE-21078, "Test Results Employed by GE for BWR Containment and Vertical Vent Loads," October 1975.

2. "General Electric ATWS Report" General Electric Response to NRC Status Report and Letter (R. E. Heineman to I. F. Stuart, April 7, 1976), June 30, 1976.