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MARK III CONTAINMENT
HYDROGEN CONTROL OWNERS GROUP

Sam H. Hobbs, Chairman

c/o Mississippi Power and Light • P.O. Box 1640 • Jackson, Mississippi 39205 601-969-2458

March 6, 1985
HGN-026

50-440/441
50-416/417
50-461/462
50-458/459

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

Attention: Mr. Robert Bernero

Dear Mr. Bernero:

Subject: Hydrogen Control Owners Group Responses to Requests for Additional Information on the CLASIX-3 Computer Code, HGN-026

Reference: Letter HGN-022 from Mr. S. H. Hobbs to Mr. H. R. Denton, dated November 7, 1984

The reference letter committed the Hydrogen Control Owners Group (HCOG) to provide generic responses to the Nuclear Regulatory Commission's (NRC) requests for additional information (RAI's) on the CLASIX-3 computer code. These RAI's were transmitted to each HCOG member individually in September, 1984. Attachment One to this letter contains the HCOG's generic responses to the NRC RAI's.

This submittal was compiled by HCOG from the best information available for submittal to the Nuclear Regulatory Commission. The submittal is believed to be complete and accurate, but it is not submitted on any specific plant docket. The information contained in this letter and its attachments should not be used for evaluation of any specific plant unless the information has been endorsed by the appropriate member utility.

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See Attached Dist

HCOG members may individually reference this letter in whole or in part as being applicable to their specific plants.

Sincerely,

S H Hobbs

S. H. Hobbs
Chairman, HCOG

SHH/mrd

Attachment

cc: Mr. Carl R. Stahle
Hydrogen Control Program Manager
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Mr. Charles G. Tinkler
Containment Systems Branch
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Mr. John Cummings, Project Manager
Hydrogen Studies, Division 4441
Sandia National Laboratory
Albuquerque, NM 87185

ATTACHMENT ONE TO HGN-026

Responses to NRC Requests for Additional Information on the CLASIX-3 Computer Code

QUESTION:

1. It is the intent of the Mark III owners to use the HCOG quarter-scale tests (which focus on diffusion-type burning within the wetwell region) and plant specific/HCOG CLASIX-3 analyses (which focus on discrete-type burning within the containment), to determine the most severe thermal environment within the containment and drywell for purposes of demonstrating equipment survivability. Since the present passive heat sink modeling in CLASIX-3 tends to underestimate the compartment atmosphere temperatures and since CLASIX-3 appears to be in non-conformance with the provisions of NUREG-0588, the CLASIX-3 containment response sensitivity studies (correspondence No. HGN-001) should not be used as the basis for determining the most severe compartment temperature conditions. In view of this concept, the present version of CLASIX-3 is inappropriate.

Since the methodology described in NUREG-0588 is generally recognized as an acceptable approach for addressing equipment qualification, describe and justify if there are deviations from the provisions of NUREG-0588 with regard to the passive heat-sink and heat-transfer assumptions that will be used for plant specific analyses in the following areas:

1) The temperature difference used with the heat-transfer film coefficients for both saturated and super-heated atmospheres;

2) The analytical model and assumptions used to account for condensate removal from the heat sink surface; and,
3) The energy removal associated with condensed mass.

RESPONSE:

Mississippi Power & Light (MP&L) Company developed a new option for heat transfer to passive heat sinks for the CLASIX-3 computer code. This option was developed in response to this request for additional information (RAI). The new heat transfer option was discussed in detail in Reference 1. The new heat transfer model represented by this option is based on a combination of the models identified in NUREG-0588, Branch Technical Position CSB 6-1 (Reference 2) and the CONTEMPT program description document (Reference 3). The model implemented in the CLASIX-3 code was developed in consultation with the NRC Containment Systems Branch Staff in order to minimize the potential for future modifications. The Hydrogen Control Owners Group intends to utilize this model to evaluate the effect of employing NUREG-0588 methodology for passive heat sink modeling.

The condensing heat transfer coefficient is based on the Uchida correlation of Reference 4. The tabular values of the coefficient as a function of the mass ratio of air to steam are presented in both the Branch Technical Position CSB 6-1 and the CONTEMPT program description document. Although the correlation is based on a mixture of air and steam, CLASIX-3 predictions of compartment atmospheres may result in vitiated air and hydrogen mixed with the steam. In determining the heat transfer coefficient, the ratio of the mass of non-condensibles to the mass of steam is used in

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CLASIX-3. The condensing region of heat transfer is defined by the wall surface temperature being below the saturation temperature corresponding to the partial pressure of water vapor in the compartment. Under these conditions, the rate of heat transfer is given by:

$$q = h_U A \, (T_s - T_w) \qquad (1)$$

where: q = rate of heat transfer

h_U = Uchida heat transfer coefficient

A = area of heat transfer

T_s = saturation temperature corresponding to the partial pressure of the water vapor

T_w = wall surface temperature

To provide a smooth transition from the condensing to superheated region, the rate of heat transfer is also evaluated at a constant value for the Uchida coefficient of 2, so that

$$q = 2A \, (T_B - T_W) \qquad (2)$$

where: T_B = bulk compartment temperature.

The largest value of q as determined by equations (1) and (2) is used.

The temperature difference in the condensing region which is used to calculate heat transfer to the passive heat sinks is either the difference between the saturation temperature and the passive heat sink surface temperature or the difference between the bulk compartment temperature and the passive heat sink surface temperature.

Under superheated conditions with the wall surface temperature above the saturation temperature, the film coefficient is calculated from the same correlation as that used in CONTEMPT. The film coefficient is given by

$$h_c = 0.13 \left[\rho_f^2 \, g \beta_f \Delta T \, c_{pf} \, k_f^2 / \mu_f \right]^{1/3} \quad (3)$$

where: g = gravitational acceleration

h_c = heat transfer coefficient

ρ_f = density of gas region

β_f = inverse of the absolute temperature of the film (assumes ideal gas)

ΔT = temperature difference between heat sink surface and bulk gas temperature

c_{pf} = specific heat of gas at constant pressure

k_f = thermal conductivity of gas region

μ_f = viscosity of gas region

The gas properties are evaluated at the average film temperature

$$(T_w + T_{bulk})/2$$

and the mass weighted average values assigned to the gas.

The heat transfer to the passive heat sinks is given by:

$$q = h_c A (T_B - T_w) \quad (4)$$

In the superheated region, the difference between bulk compartment gas temperature and the heat sink surface temperature is used to calculate heat transfer to the heat sink.

The model and assumptions used to account for condensate removal are consistent with NUREG-0588. NUREG-0588 specifies that 92% of the condensing heat transfer is assumed to be derived from condensation and 8% is assumed to be removed directly from the bulk compartment atmosphere. The rate of condensation is:

$$\dot{m}_u = 0.92\, q/(h_B - h_f) \qquad (5)$$

where: $\dot{m}_u$ = rate of condensation
h_B = bulk enthalpy of vapor
h_f = saturated liquid enthalpy corresponding to T_s

The condensate is assumed to be immediately removed to the sump so that there is no revaporization of condensate from the walls.

The energy removed by condensation is given by

$$q_{rem} = \dot{m}_U\, (h_B - h_f) \quad (6)$$

The HCOG will utilize this heat transfer model in subsequent generic CLASIX-3 analyses of the drywell response to degraded core accidents initiated by a small break accident. The HCOG's program for analyzing the drywell response is discussed in Task 10 of the HCOG Hydrogen Control Program Plan (transmitted to the NRC in Reference 5). This heat transfer option is not expected to have any significant effect on the containment, wetwell or drywell response to a degraded core accident initiated by a stuck open relief valve. The HCOG will complete one additional CLASIX-3 analysis utilizing the new option for heat transfer with the stuck open relief valve base case assumptions identified in the HCOG CLASIX-3 Sensitivity Analysis (Reference 6). The results from the new run will be compared with the results from the CLASIX-3 stuck open relief valve base case to demonstrate that the new heat transfer option does not significantly affect the containment, wetwell, or drywell response to a degraded core accident initiated by a stuck open relief valve.

References

1. Letter AECM-83/0455 dated August 13, 1983 from Mr. L. F. Dale to Mr. H. R. Denton.

2. "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation", Branch Technical Position CSB 6-1, dated July 1981.

3. "CONTEMPT-LT-A Computer Program for Predicting Containment Pressure - Temperature Response to a Loss of Coolant Accident", AWCR-1219, UC-78, dated June 1975.

4. H. Uchida, A. Oyama, and T. Toga, "Evaluation of Post Incident Cooling Systems of Light Water Power Reactors", Proceedings Third International Conference on the Peaceful Uses of Atomic Energy, Volume 13, Session 3.9, United Nations, Geneva (1964).

5. HGN-024 letter from S. Hobbs to H. Denton dated December 14, 1984.

6. CLASIX-3 Containment Response Sensitivity Studies.

QUESTION:

2. For each postulated degraded core sequence, (i.e., SORV and drywell break initiated events), provide an evaluation of the impact on the drywell atmosphere environment when considering heat losses from the reactor vessel and its associated piping (e.g., SRV lines). Provide and justify assumptions used in your evaluation, e.g., convective and radiative heat transfer parameters.

RESPONSE:

The impact on the drywell environment by considering heat losses from the reactor vessel and its associated piping for degraded core accidents initiated by stuck open relief valves is negligible in comparison with the impact of these effects from accidents initiated by small break accidents in the drywell. Degraded core accidents initiated by small breaks in the drywell will establish the limiting thermal environment for equipment survivability in the drywell. The HCOG will consider the heat loads from the reactor vessel and its associated piping in the analysis of the drywell response to degraded core accidents which will be conducted as part of Task 10 in the Hydrogen Control Program Plan which was transmitted to the NRC in reference 1.

The drywell response in degraded core accidents initiated by stuck open relief valves is not a limiting thermal environment for equipment survivability. The heat losses from the reactor vessel and associated piping does not need to be considered in the calculation of drywell response to degraded core accidents initiated by a stuck open relief valve.

Reference

1. HGN-024 letter dated December 14, 1984 from Mr. S. H. Hobbs to Mr. H. R. Denton.

QUESTION:

3. According to the BWR/6 Standard Technical Specifications, periodic low pressure leak testing of the drywell is required. The acceptance criterion is that the leakage shall be less than or equal to 10% of the maximum $A/\sqrt{K}$ (i.e., approximately 1 ft^2). Thus, the maximum allowable leak rate is equivalent to roughly 4000 SCFM at 3 psi pressure differential. Provide an evaluation of the consequences within the drywell and the containment by the combustion of hydrogen when considering the drywell bypass leakage (include mechanistically the effects of upper pool dump and pool drawdown).

RESPONSE:

Requirements were established to consider possible leakage from the drywell to the containment which bypasses the suppression pool. Each Mark III containment plant has established the plant's capability to withstand the effects of suppression pool bypass leakage considering the operation of engineered safeguard feature containment heat removal systems. Technical specification requirements for verifying that the leakage shall be less than or equal to 10% of the allowable $A/\sqrt{K}$ were established non-mechanistically. These requirements were imposed to assure that the actual suppression pool bypass always remains below the capability value for the life of the plant.

HCOG recognizes that suppresion pool bypass leakage may affect the drywell and containment response to accidents initiated by small breaks in the reactor coolant pressure boundary piping inside the drywell. Accordingly, HCOG will consider the effects of suppression pool bypass as part of

Task 10, Evaluation of Drywell Response to Degraded Core Accidents in the Hydrogen Control Program Plan submitted by reference 1.

<u>Reference</u>

1. HGN-024 letter dated December 14, 1984 from Mr. S. H. Hobbs to Mr. H. R. Denton.