Docket No. 50-458

SEP 0 6 1984

Mr. William J. Cahill, Jr.
Senior Vice President
River Bend Nuclear Group
Gulf States Utilities Company
Post Office Box 2951
Beaumont, Texas 77704
ATTN: J. E. Booker

Dear Mr. Cahill:

SUBJECT: REQUEST FOR ADDITION INFORMATION - HYDROGEN CONTROL FOR MARK III CONTAINMENTS

As a part of the staff's continuing review of hydrogen control for Mark III containments during postulated degraded core accidents, the staff has identified the need for additional information on the CLASIX-3 code which has been used to support the licensing activities associated with Mark III plants. The CLASIX-3 code has been used to determine the environmental conditions to which equipment survivability is to be evaluated. This request for information is included in the enclosure.

Please inform NRC Project Manager, Edward Weinkam of your schedule for response and for clarification or further discussion on this topic.

Sincerely,

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclsoure: As stated

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REQUEST FOR ADDITIONAL INFORMATION RELATED
TO DEGRADED CORE HYDROGEN CONTROL

1. It is the intent of the Mark III owners to use the HCOG quarter-scale tests (which focuses on diffusion-type burning within the wetwell region) and plant specific/HCOG CLASIX-3 analyses (which focuses 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 concern, 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.

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.

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 allowable $A/\sqrt{K}$ (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).

River Bend Station

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