Department of Energy
Washington, D.C. 20545

Docket No. 50-537
HQ:S:82:161

DEC 22 1982

Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Check:

RESPONSE TO THE NUCLEAR REGULATORY COMMISSION QUESTION OF OCTOBER 12, 1982

Reference: Letter, P. S. Check to J. R. Longenecker, "CRBRP, Request for Additional Information," dated October 12, 1982

Enclosed is a response to a request for additional information identified in the reference letter concerning the Intermediate Heat Transport System tee of the Clinch River Breeder Reactor plant.

Any questions regarding the information can be addressed to D. Robinson (FTS 626-6098) of the Project Office Oak Ridge staff.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of
Breeder Demonstration Projects
Office of Nuclear Energy

Enclosure

cc: Service List
Standard Distribution
Licensing Distribution

D001
1/40

Enclosure

NRC Question of October 12, 1982:

In the past there has been some concern shown by the Project for the IHTS tee which connects the SWRS rupture discs and associated systems to the IHTS evaporator inlets. The concern in question was the possibility that in the course of a large or intermediate sodium/water reaction a "standing reaction front" might develop at the junction of the tee for a period of time sufficient to overtemperature the tee. The situation has been logically named the "hot tee" problem.

Please provide the current status on the "hot tee" problem. If the Project no longer considers it a problem, please provide the analysis upon which that conclusion is based.

Response:

Analyses of the sodium/water reaction (SWR) design basis leak (DBL) have been performed in order to predict reaction zone/sodium interface histories and flow and, pressure and fluid temperature histories in the intermediate heat transport system (IHTS) and in the sodium/water reaction pressure relief subsystem (SWRPRS). Additional analyses of the SWR DBL have been performed assuming failure of an evaporator or superheater inlet isolation valve to close. These analyses do not indicate a structural integrity problem at the IHTS tee during the SWR DBL with or without failure of an isolation valve. A discussion of the computer codes utilized and the predicted plant performance during these events is provided in the subsequent paragraphs.

Results from the Transient Sodium/Water Reaction Analysis Program[1] (TRANSWRAP) for SWR DBLs located at the evaporator upper tubesheet and in the superheater at the elevation of the superheater upper sodium nozzle have been input to an updated version of the Hydraulic Intermediate System Transient Analysis Method[2] (HISTAM) code. TRANSWRAP models the initial few seconds and HISTAM models the longer-term progression of the event. Detailed descriptions of the TRANSWRAP code and models are included in Reference 3. The HISTAM code models the entire sodium-side of the IHTS loop with SWRPRS and includes:

- hydrogen bubble generation as the SWR proceeds;
- pump tank and expansion tank cover gas spaces;
- pump coastdown;
- the SWRPRS through and including the reaction products separator tank;
- evaporator and superheater vent lines to expansion tank;
- two-phase (hydrogen/sodium) flow option in the SWRPRS relief line attached to the faulted evaporator or superheater; and

- calculation of fluid bulk temperatures in the IHTS tee assuming an adiabatic, homogenous mix at thermodynamic equilibrium.

The RELAP4MOD5[4] code has been used to model the water-side of the faulted evaporator or superheater in order to produce water injection histories. The RELAP code has been incorporated into the SWR design methodology because of its widespread use and extensive verification in light water reactor safety analysis. RELAP can be used to model subcooled water, steam, and two-phase steam/water transient flows for a variety of initial and boundary conditions including heat transfer. The RELAP models included the faulted evaporator or superheater, the steam drum, the relief valves, the evaporator water dump valves, the isolation valves, and the three equivalent double-ended guillotine tube breaks which constitute the DBL. Existing RELAP results for the SWR DBL case were extrapolated by hand in order to provide water injection histories for the extended duration HISTAM analyses.

Results of the SWR DBL events with assumed failure of an isolation valve are discussed below.

SWR DBL in Superheater, Superheater Inlet Isolation Valve Fails to Close

Within two seconds into the transient, the SWR hydrogen bubble has reversed the sodium flow in the superheater upper sodium nozzle, and the reaction zone moves to the region of the adjacent IHTS tee. The tee is fed with a decelerating flow of hot leg sodium as the IHTS pump coasts down and with a decelerating flow of steam as the steam drum and superheater depressurize through the superheater safety/relief valves. The sodium flow into the tee is predicted to pass through the stoichiometric level at 19.5 seconds. During the short duration of relatively near stoichiometric sodium flow (three seconds) a localized fluid temperature in the region of the tee of 1500 to 2000°F can be predicted. The short duration elevated temperature does not indicate a potential problem of tee structural integrity.

SWR DBL in Evaporator, Evaporator Inlet Isolation Valve Fails to Close

Within three seconds into the transient, the faulted evaporator has essentially been cleared of sodium, and the hydrogen bubble has also cleared the adjacent IHTS tee. Within four seconds the faulted evaporator SWRPRS relief line has cleared of liquid sodium, the bubble pressure has collapsed, and the sodium flow beyond the tee has reversed due to the head in the pump and expansion tanks. At ten seconds, the bubble/sodium interface has returned to the region of the tee. Sodium flow into the tee ranges between two-to-three times stoichiometric through twenty seconds resulting in a localized fluid temperature of 1500 to 2000°F, thereafter dropping as the stoichiometric ratio

increases. Pressure in the tee does not exceed 46 PSIA during the period of elevated temperature. The short duration elevated temperature does not indicate a potential problem of tee structural integrity.

The conclusion is that analyses do not indicate a structural integrity problem at the IHTS tees during SWR DBLs with assumed failure of an isolation valve.

REFERENCES

1) Knittle, D. E., "TRANSWRAP II User's Manual," dated February 1981 and Knittle, D. E., "TRANSWRAP II Problem Definition Manual," dated February 1981 (both transmitted by GE letter XL-611-10011, dated February 20, 1981; and C. R. Bell "TRANSWRAP - A Code for Analyzing the System Effects of Large-Leak Sodium-Water Reactions in LMFBR Steam Generators," Proc. ANS Conf. Fast Reactor Safety, Beverly Hills, USAEC-Conf-740401 (1974); and Atomics International Supporting Document TI-001-130-025, "TRANSWRAP - A Compressible Hydrodynamic Code for Large-Leak Sodium/Water Reaction Analysis," dated February 5, 1973.

2) J. W. Whalen and T. Rauch, "Hydraulic Intermediate System Transient Analysis Method HISTAM-2 Computer Code," CRP-004, General Electric Company Fast Breeer Reactor Department, Sunnyvale, California, July 1976.

3) Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Chapter 5, Section 5.5.3.6.2, Page 5.5-28, Amendment 62, November 1981.

4) "RELAP4/MOD5 A Computer Progrm for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems," Volumes I and II, prepared by Aerojet Nuclear Company for the U.S. NRC and ERDA under Contract E (10-1)-1375, ANCR-NEREG-1335, September 1976.