NRC PDR

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

MAY 5 1980

Dr. Milton S. Plesset, Chairman
Advisory Committee on Reactor Safeguards
Washington, D. C. 20555

Dear Dr. Plesset:

I am responding to your letter of March 11, 1980 in which the ACRS recommended that consideration be given, if at all possible, to some verification testing of the "feed-bleed" and/or "reflux-condensation" processes during the low power testing program for the near term OL plants. This topic was partially discussed with the ACRS Ad-Hoc Subcommittee on Natural Circulation Heat Removal on March 26, 1980. We plan further discussions with the ACRS Subcommittee on the Sequoyah plant during the next few months. The enclosure presents the staff views of the practicality of performing verification testing on a power reactor for both the "feed-bleed" process and the "reflux-condensation" process.

For the reasons cited, the NRC staff does not consider it prudent to attempt demonstrations of either the "feed-bleed" or the "reflux-condensation" process in a power reactor as part of the low power test program.

Despite this view, staff is in general agreement with the need to develop a coordinated effort to evaluate shutdown heat removal requirements in a comprehensive manner (ACRS letter of April 17, 1980 on TMI Action Plan). The final version of the Action Plan will include an item calling for the ACRS and the staff to work together to develop plans for such a coordinated effort.

Sincerely,

(Signed) William J. Dircks

William J. Dircks
Acting Executive Director for Operations

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ENCLOSURE

"Feed-Bleed" Process

The "feed-bleed" process is one that may be used in an operating plant to remove decay heat in the event all feedwater is lost. The process consists of adding inventory to the reactor coolant system (RCS) by the high pressure safety injection (HPI) pumps and discharging fluid from the RCS through the power operated relief valves (PORV) and/or the safety valves. The "feed-bleed" process would be used only if there were no main or auxiliary feedwater available to the shell side of the steam generators. There is currently no NRC requirement that a nuclear plant be designed with a capability for "feed-bleed" operation.

The ability of the "feed-bleed" process to sucessfully provide core cooling is dependent on the relieving capacity of the PORV's, the shutoff head of the HPI's, and the overall system transient response, including the time at which the PORV's are opened. In many plants the use of the "feed-bleed" process will require that the PORV's be opened and the RCS pressure be reduced to enable the HPSI pumps to replenish the RCS inventory at an acceptable rate.

The analytical tools currently available are considered adequate to perform an analysis of the "feed-bleed" process, although no experimental information is currently available to verify this adequacy. The performance of such an analysis is hampered because the discharge characteristics of the PORV's with single phase liquid flow and with two-phase steam-water flow are not currently well known. As a part of the short-term lessons learned requirements (NUREG-0578), PORV and safety valves will be tested to obtain experimental data regarding these flow characteristics. Once this is accomplished, the "feed-bleed" capability can be more accurately evaluated.

To demonstrate the "feed-bleed" process as part of a near term operating license low power test program is complex since fission heat must be used to stimulate decay heat. This must be done with boiling in the reactor core coolant. Thus, reactivity control must be maintained with steam voids in the reactor coolant. As the system is depressurized, the steam voids will increase and the control assemblies must be further withdrawn to maintain the desired power level. If there were an equipment malfunction or operator error that caused a flow of relatively cool water from the steam generator tubes and cold legs into the core, a significant reactivity transient could occur.

There are other potential difficulties with the demonstration of the "bleed-feed" process. One of these deals with the collection of the discharge from the RCS through the PORV's and the safety valves. Although this fluid flows to the quench tank, neither the volume nor the design pressure of this tank is sufficient to run a demonstration test. Therefore, additional systems would have to be provided to cool and collect the discharge from the quench tank. Both the cooling and collection would be difficult in a power reactor that has been already constructed. Because of space constraints, it may be necessary to pipe the fluid through a spare containment penetration (assuming one is available) to equipment outside of containment. The design criteria and detailed design of a cooling and collection system located outside of containment would have to be thoroughly reviewed to determine its acceptability from the viewpoint of public risk.

The fact that the reactor coolant would be boiling at high pressure during the demonstration is another concern. Since the test would be conducted with a new core, the reactor coolant would have to be heavily borated for reactivity control. The boiling process may cause some plateout of boron on the fuel cladding and possibly adversely affect the useful life of the core.

Because of the above difficulties, the Semiscale facility seems more appropriate for a "feed and bleed" experiment than a commercial power reactor. A test in the Semiscale facility is planned late April of CY 80 in which all feedwater will be terminated. While the primary purpose of this test is to verify analytical models regarding capability to calculate steam generator dryout time, PORV and/or safety valve lift time, and primary system inventory depletion rate, we are presently examining the capability of recovering the system inventory during the test solely with the use of HPI pumps. We believe this can be successfully accomplished while the valve discharge is two-phase. Once the discharge is low quality two-phase or single-phase liquid, mass storage becomes a concern. We will keep you advised of our plans for this test as they are evolved.

Reflux-Condensation

We interpret the term "reflux-condensation" as any heat transport mechanism that involves two-phase reactor coolant conditions in the core region and condensation in the steam generator. This process will occur following a loss-of-coolant accident having a break size that results in (1) a loss of inventory from the RCS great enough so that single phase liquid natural circulation within the reactor coolant system is not maintained, and (2) insufficient energy removal by inventory loss through the break to remove from the RCS all of the decay heat being generated. Therefore, the steam generators must be relied upon to remove some fraction of the decay heat.

The "reflux-condensation" could actually involve several modes of heat transport. One mode would be a two-phase mixture that is transported from the core to the steam generator by natural convection. A second mode could be transport of steam to the steam generator through the hot leg and return of the condensate to the reactor vessel through the cold leg by gravity flow. A third mode could be a true reflux condensing process in which the condensate flows counter-current to the direction of steam flow and returns to the reactor through the hot leg. In addition, there may be combinations of the above modes, particularly in a U-tube steam generator where both the second and third modes described above could occur simultaneously in different steam generator tubes.

We have evaluated these various modes of heat transport and has concluded that they all provide acceptable means of cooling the core following a small loss-of-coolant accident. This evaluation included consideration of non-condensable gases in the RCS from dissolved hydrogen in the primary coolant, dissolved nitrogen in the accumulator water, dissolved air in the refueling water storage tank, free nitrogen used to pressurize the accumulators, hydrogen released from radiolytic decomposition of injected water, fission and fill gas in reactor fuel, hydrogen gas (free and dissolved) in the makeup tank, and pressurizer steam space gas.

Despite the positive conclusion reached during its evaluation, the staff believes it is prudent to have experimental verification of the various modes of natural circulation, including both fluid flow and condensation heat transfer. An extensive test program to be performed at a variety of experimental facilities is in the planning stage. The experimantal program plan includes examining the effect of non-condensable gases on two-phase natural circulation. The status of these plans were described by the staff at the ACRS Ad-Hoc Subcommittee on Natural Circulation Heat Removal during the March 26, 1980 meeting of that subcommittee.

In addition, the B&O Task Force has recommended that each vendor provide verification of his analytical models used to calculate the various modes of two-phase natural circulation predicted in his plant.

To perform a demonstration of the "reflux-condensation" process in a power reactor, the water level in the vessel and the steam generator would have to be lowered to the height of the reactor coolant nozzles. The pressurizer, in a Westinghouse or Combustion Engineering designed reactor, would drain naturally into the hot leg and would be empty. Like a demonstration of the "feed-bleed" process, decay heat in the fresh core would have to be simulated using fission heat while the core is cooled by a two-phase mixture of water and steam. There would be a potential for reactivity excursions due to equipment malfunction and/or operator error. With an empty pressurizer, the only means of RCS pressure control would be by adjusting the heat input from the core and the heat removed by the condensing process. Changes in pressure would cause changes of the steam volume fraction in the core, making reactivity control very difficult.

Even such a demonstration would not answer questions of the effect of non-condensable gases. If gases were added to the RCS as part of the reactor demonstration of the "reflux-condensation" process, this would lead to changes in system pressure and possible different heat transfer rates in the steam generator. This would make reactivity control even more difficult. Moreover, it would not be possible to measure the distribution and concentration of the gas within the primary system.