NRC PDR

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

JUN 4 1979

Mr. Tom M. Anderson, Manager
Nuclear Safety Department
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Anderson:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING SMALL BREAK LOCA ANALYSIS

As you are aware, the NRC staff is evaluating certain aspects of the NRC licensing process in light of the Three Mile Island Unit 2 (TMI-2) accident, of March 28, 1979. One of the aspects being addressed in this evaluation is the status of models used in safety analyses, especially those used to analyze transients and small break LOCA's. In order for the staff to complete its evaluation of the response of currently operating Westinghouse-designed plants to postulated small break LOCA's, additional information is required. These information needs, which pertain to expected system behavior following postulated small break LOCA's and small break LOCA analysis methods and results, are identified in the enclosure to this letter.

Please provide, within seven days of the date of receipt of this letter, your schedule for responding to the items contained in the enclosure. We recognize that some of the requested information may have already been provided to the staff in the course of other reviews. In lieu of referencing already submitted material, we would prefer, in order to facilitate our review, that the responses be provided in one complete document.

The information contained in the enclosure was discussed with your representatives at our May 31, 1979 meeting. If you have any question regarding the contents of this letter, please contact P. D. O'Reilly, the assigned project manager.

Sincerely,

D. F. Ross, Jr., Deputy Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosure:
As stated

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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION
REGARDING SMALL BREAK LOCA ANALYSIS

The response of the primary system of a given plant to small break LOCA's will differ greatly depending upon the break size, the location of the break, mode of operation of the reactor coolant pumps, numbers of ECCS systems functioning, and the availability of secondary side cooling. In addition, this response may differ for different plants designed by the same NSSS vendor because of differences in loop configuration (two-loop, three-loop, or four-loop) or different ECCS designs. In order for the staff to complete its evaluation of the response of currently operating Westinghouse PWR designs to postulated small break LOCA's, the following information is needed:

(1) Provide a qualitative description of expected system behavior for (a) a range of postulated small break LOCA's, including the zero break case, and (b) feedwater-related limiting transients combined with a stuck-open power operated relief valve. These cases should include situations where auxiliary feedwater is both assumed available and not available. The cases considered should also include breaks large enough to (a) depressurize the primary system, (b) maintain the primary system at some intermediate pressure, and (c) repressurize the primary system to the safety and/or relief valve setpoint pressure. Various break locations in the primary system should be considered, including the pressurizer.

(2) Provide a qualitative description of the various natural circulation modes of expected system behavior following a small break LOCA. Discuss any ways in which natural circulation can be interrupted. In particular, discuss the applicability of the concerns in the Michelson reports (reports on B&W 205 FA plants and CE System 80 plants) identified in Annex 1 to this Enclosure. Assess the possible effects of non-condensible gases contained in the primary system.

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The following questions pertain to your small break LOCA analysis methods:

(3) Demonstrate that your current small break LOCA analysis methods are appropriate for application to each of the cases identified in Items (8) through (12) below. This demonstration should include an assessment of the adequacy of the pressurizer and steam generator noding, and the pressurizer surge line representation. This may be accomplished by verifying the methods with the use of data (e.g., comparison with experiments, TMI-2 evaluation).

If, as a result of the above assessment, you modify your analysis methods (e.g., pressurizer and steam generator noding), provide justification for any such modification.

(4) Verify the break flow model used for each break flow location analyzed in the response to Item (8) below.

(5) Verify the analytical model used to calculate natural circulation heat removal under two-phase flow conditions.

(6) Provide justification for your treatment of non-condensible gases following discharge of the safety injection tanks.

(7) Verify your analytical calculation of fluid level in the reactor pressure vessel for small break LOCA's and feedwater transients.

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For each of the analyses requested in Items (8) through (12) below,

(i) Provide plots of the output parameters specified in Annex 2 to this Enclosure.

(ii) Indicate when the pressurizer safety and/or relief valves would open.

(iii) Include appropriate information about the role of control systems in the course of the transient. Describe how the system response would be affected by control systems.

(iv) If the scenario is different for different classes of plants (two-loop, three-loop, four-loop, different ECCS designs), provide an example of each kind.

(8) Provide the results of a sample analysis of each type of small break behavior discussed in the response to item (1) (e.g., depressurization, pressure hangup, repressurization).

(9) Provide the results of an analysis of the worst break size and location in terms of core uncovering. This may be a break which does not result in HPI initiation. This may require more than one calculation.

(10) Provide the results of a complete analysis of feedwater-related limiting transients combined with a stuck-open power operated relief valve. These cases should include situations where auxiliary feedwater is both assumed available and not available.

(11) Provide the results of a small break LOCA analysis assuming loss of feedwater and auxiliary feedwater. The case with the worst break location which affords the least amount of time for operator action should be analyzed. Single failure of the ECCS should be considered.

(12) Provide the results of a small break LOCA analysis assuming that one steam generator is lost either due to isolation or due to loss of auxiliary feedwater.

(13) Provide the results of an analysis of the effect of reactor coolant pump operation (tripping all RCP's, keeping all and some RCP's running) on the course of small break LOCA's.

(14) Provide the results of an analysis of the effects of different HPI termination criteria on the course of small LOCA's. Specifically, for each small break LOCA analyzed in response to Item (8) above, compare the effects of the NRC HPI termination criteria (as stated in I&E Bulletins 79-06A and 79-06A, Rev. 1, Item 7(b)) to those for the HPI termination criteria which you have recommended to licensees with Westinghouse designed operating plants. Provide plots of significant parameters of interest, such as system pressures, temperatures, and subcooling, on a common time axis. Indicate on the plot when the operator would terminate HPI injection for both sets of criteria.

(15) Provide a list of transients expected to lift the PORVs; identify the assumed steam and two-phase flow rates through the valves for these transients. Provide justification for your assumptions, including the time at which two-phase flow discharge would be experienced.

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(16) Provide guidelines for the preparation of operational procedures for the recovery of plants following small LOCA's. This should include both short-term and long-term situations and follow through to a stable condition. The guidelines should include recognition of the event, precautions, actions, and prohibited actions.

If RC pump operation is assumed under two-phase conditions, a justification of pump operability should be provided. Discuss instrumentation available to the operator and any instrumentation that might not be relied upon during these events (e.g., pressurizer level). What would be the effect of this instrumentation on automatic protection actions?

ANNEX 1

TVA (C. Michelson) Concerns

1. Pressurizer level is an incorrect measure of primary coolant inventory.

2. The isolation of small breaks (e.g., letdown line; PORV) not addressed or analyzed.

3. Pressure boundary damage due to loadings from a) bubble collapse in subcooled liquid and 2) injection of ECC water in steam-filled pipes.

4. In determining need for steam generators to remove decay heat, consider that break flow enthalpy is not core exit enthalpy.

5. Are sources of auxiliary feedwater adequate in the event of a delay in cooldown subsequent to a small LOCA?

6. Is the recirculation mode of operation of the HPSI pumps at high pressure an established design requirement?

7. Are the HPSI pumps and RHR pumps run simultaneously? Do they share common piping?/suction? If so, is the system properly designed to accommodate this mode of operation (i.e., are any NPSH requirements violated, etc...?)

8. Mechanical effects of slug flow on steam generator tubes needs to be addressed. (transitioning from solid natural circulation to reflux boiling and back to solid natural circulation may cause slug flow in the hot leg pipes).

9. Is there minimum flow protection for the HPSI pumps during the recirculating mode of operation?

10. The effect of the accumulators dumping during small break LOCAs is not taken into account.

11. What is the impact of continued running of the RC pumps during a small LOCA?

12. During a small break LOCA in which offsite power is lost, the possibility and impact of pump seal damage and leakage has not been evaluated or analyzed.

13. During transitioning from solid natural circulation to reflux boiling and back again, the vessel level will be unknown to the operators, and emergency procedures and operator training may be inadequate. This needs to be addressed and evaluated.

NOTE: Items 1 through 4 are taken from "Decay Heat Removal During A Very Small Break LOCA for a B&W 205-Fuel Assembly PWR," C. Michelson, Draft Report, January 1978.

Items 5 through 15 are taken from "Decay Heat Removal Problem Associated with Recovery from a Very Small Break LOCA for CE System 80 PWR," C. Michelson, Draft Report, May 1977.

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'. The effect of non-condensible gas accumulation in the steam generators and its possible disruption of decay heat removal by natural circulation needs to be addressed.

r. Delayed cooldown following a small break LOCA could raise the containment pressure and activate the containment spray system. Impact and consequences need addressing.

ANNEX 2

Plotted Output Parameters

Core: $\underline{L}$, $\underline{X}$, $\underline{T_{CL}}$

Reactor Vessel:

Upper Head: L, X

Downcomer: L, X

Piping:

Hot Leg: $\underline{X}$, $\underline{T}$, W, L (Pressurizer Leg)

Cold Leg: $\underline{X}$, $\underline{T}$, W, L, W_{HPI}, $\int W_{HPI} dt$ (Break Leg)

Pressurizer: Win, Xin, $\underline{L}$, $\underline{X}$, $\underline{P}$, T

Steam Generator:

Primary: X, L, T, h

Secondary: P, $\underline{L}$, X, T, W_{REL}, W_{AFW}, h

Leak:

PORV, W, X

or

Break, $\underline{W}$, $\underline{X}$, $\int Wdt$

Pump Loop Seal: X, L

Nomenclative: P - Pressure
L- Mixture Level
X - Quality
T - Temperature
W - Mass Flow Rate

h - film heat transfer coefficient
HPI - High Pressure Injection
REL - Relief Valve
AFW - Auxiliary Feedwater