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SUBJECT:
RESPONSE TO NRC"S REQUEST OF 12/16/77...FURNISHING INFO CONCERNING PERMANENT RESOULTION, AND SCHEDULE FOR IMPLEMENTATION, TO STAFF CONCERNS ABOUT UPPER PLENUM INJECTION OF EMERGENCY CORE COOLING SYSTEM WATER.

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LEON D. WHITE, JR.
VICE PRESIDENT

TELEPHONE
AREA CODE 716 546-2700

March 15, 1978

Director of Nuclear Reactor Regulation
ATTN: Mr. Dennis L. Ziemann, Chief
Operating Reactor Branch #2
U.S. Nuclear Regulatory Commission
Washington, DC 20555

50-244

Dear Mr. Ziemann:

A letter from Mr. Edson G. Case dated December 16, 1977 requested that we provide a permanent resolution, and a schedule for its implementation, to staff concerns about upper plenum injection (UPI) of emergency core cooling system (ECCS) water. We believe that the models currently used to evaluate ECCS performance are conservative and comply with Appendix K to 10 CFR Part 50. We will, however, take the action outlined below in order to satisfy the staff requests in the letter of December 16, 1977.

A new ECCS evaluation model will be developed to address the Safety Evaluation Report[1] (SER) issued by the NRC on December 16, 1977. This SER requested that explicit accounting of upper plenum injected water be simulated during a loss of coolant accident. Previous ECCS analyses have assumed that UPI water adds directly to bottom reflooding with no accounting of steam-water interaction. Such a model is conservative since no credit is taken for steam condensation and fuel rod cooling as the UPI water penetrates the core. However, in an effort to be responsive to the SER, we have developed a work scope which will include steam-water interaction due to upper plenum injection during the reflood portion of the accident. The work is segregated into three phases: development of model, generic sensitivity study and individual plant analysis.

The development of the new model will use as a starting point the Westinghouse evaluation codes WREFLOOD[2] and LOCTA[3]. Added to these appropriate codes will be an average model approach for simulating the UPI water and its interaction with the system. The approach for handling the UPI water interaction is similar to the approach used by the NRC in its SER and in our submittals of January 16, and February 15, 1978. The simulation of the UPI water will be accounted for during the reflood portion of the transient and no blowdown effect will be considered.

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The basis for eliminating modeling of the UPI water during the blowdown portion of the transient is based upon two generic 2-loop sensitivity studies[4,5]. These studies showed that for all postulated breaks from a 3 ft^2 split break to the full double ended cold leg guillotine (DECLG) (C_D - 1.0, .8, .6, .4) safety injection does not occur during blowdown. Therefore, no modification to the SATAN-VI code used during the blowdown portion of the transient is required. The DECLG breaks have been shown to be the most limiting breaks. For small breaks, the results show that the core pressure is above the shutoff head of the low head safety injection pumps (UPI) for 8 inch diameter breaks and smaller. Therefore, no UPI will occur and it need not be modeled. For breaks between 8 inches and 3 ft^2 splits UPI will occur during the blowdown portion of the transient. However, for such breaks the core recovers very quickly (> 1 ft/sec) due to accumulator injection. Assuming the steam-water interaction model yields about the same flooding rate change as for the large breaks ($\pm$0.2 in/sec), the relative effect is much smaller for these breaks than the limiting breaks. As indicated in our January and February, 1978 submittals there is little effect on peak clad temperature for the limiting breaks. Therefore, since the small breaks have 500-700°F margin to the acceptance criterion, no modification will be made to either WFLASH or SATAN-VI to run breaks sizes between 8 inches and 3 ft^2.

In the new model the following effects will be considered; metal heat in the upper plenum, top quench front propagation, vaporization of UPI water, horizontal and vertical entrainment, decay heat in the fuel and integration with the existing WREFLOOD code to get the overall system and feedback effects. Upon completion of the development phase, generic sensitivity studies will be performed including a large break spectrum for double-ended cold leg guillotine ruptures (C_D = 1.0, .8, .6, .4). Single failure criteria will be examined assuming loss of offsite power and no pumped SI during blowdown.

Upon completion of the sensitivity studies, a worst break analysis will be performed for our plant using the limiting conditions determined by the generic sensitivity study.

For the work scope described above the estimated calendar time for completion is:

Development	5-6 months
Sensitivity	2 months
Plant Analysis	1 month
Report	1 month
	9-10 months

This schedule is an estimate of the time required to accomplish the basic scope of work and does not include the time required for NRC staff review or the time required to respond to requests for additional information.

It will be desirable that the NRC staff review and approve the model outlined above prior to the time that sensitivity studies or specific plant analyses are performed. We will therefore contact you to arrange a meeting to discuss our results as we near the end of the model development phase of our work. If you have questions which will affect the action outlined above please contact us prior to that time.

Sincerely yours,

L D White, Jr.

L.D. White, Jr.

LDW:cem

Attachment: References

REFERENCES

1. Case, E.G., Letter, dated December 16, 1977, "Safety Evaluation Report on ECCS Evaluation Model for Westinghouse Two-Loop Plants".

2. Collier, G., et al "Calculational Model for Core Reflooding often a "Loss of Coolant Accident (WREFLOOD Code)" WCAP-8170, June 1974.

3. Bordelon, et al, "LOCTA-IV Program: Loss of Coolant Transient Analysis", WCAP-8301-June 1974.

4. Delsignore, et al, "Westinghouse ECCS Two-Loop Plant Sensitivity Studies (14 x 14)", WCAP-8854-A, May 1977.

5. "Westinghouse Emergency Core Cooling System - Plant Sensitivity Studies", WCAP-8340, July 1974.