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November 15, 1982

SBN-370
T.F. B7.1.2

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. George W. Knighton, Chief
Licensing Branch 3
Division of Licensing

References:
(a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) USNRC Letter, dated February 12, 1982, "Request for Additional Information," F. J. Miraglia to W. C. Tallman
(c) PSNH Letter, dated March 12, 1982, "Response to 440 Series RAIs; (Reactor Systems Branch)," J. DeVincentis to F. J. Miraglia

Subject: Revised Response to RAIs 440.22, 440.45, and 440.52

Dear Sir:

We have enclosed revised responses to the subject Requests for Additional Information which were forwarded in Reference (b).

The original responses to these RAIs were submitted in Reference (c).

The enclosed responses will be included in OL Application Amendment 48.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

J. DeVincentis
Project Manager

ALL/fsf

B001

440.22 Provide the following information related to pipe break or leaks in high or moderate energy lines outside containment associated with the RHR System when the plant is in a shutdown cooling mode.

1. Determine the maximum discharge rate from a pipe break in the systems outside containment used to maintain core cooling.

2. Determine the time available for recovery based on these discharged rates and their effect on core cooling.

3. Describe the alarms available to alert the operator to the event, the recovery procedures to be utilized by the operator, and the time available for operator action.

A single failure criterion consistent with Standard Review Plan 3.6.1 and Branch Technical Position APCSB 3-1 should be applied in the evaluation of the recovery procedures utilized.

RESPONSE: While in the shutdown cooling mode, only the RHR and CC Systems are utilized to maintain reactor core cooling. Because the RHR System operates as a "high energy" system for less than 2% of the time which it operates, it is considered a "moderate energy" system in accordance with BTP ASB 3-1, Appendix A - Definitions.

1. The maximum discharge rate due to a moderate energy line break of the above two systems is as follows:

   a) For RHR lines RC-13-2-601-12" and RC-58-2-601-12" (see Note below), the maximum discharge rate is 121.2 cfm.

   b) For CC lines CC-775-5-152-20" on CC-829-2-152-20", the maximum discharge rate is 85.4 cfm.

   NOTE: Cracks in the 14" and 16" sections of the RHR suction lines are not postulated because maximum stresses in these lines are less than those specified in BTP MEB 3-1, Section B.2.c.(1).

2. In determining the time available for recovery, a conservative estimate was made as to how much RCS inventory would have to be lost before the RHR System would loose its ability to perform core cooling. Although the RHR System will continue to perform its function with the RCS water level as low as the centerline of the vessel loop nozzles, only the inventory associated with 1) the pressurizer (at 25% level), 2) the surge line, 3) the primary side of the steam generators and 4) reactor vessel above the loop nozzles was considered. This inventory conservatively amounts to more than 5840 cu. ft.

   Assuming that a loss of 5840 cu. ft. would begin to degrade the RHR System's ability to maintain core cooling, and neglecting any mass input associated with the Charging System or operator action associated with manually initiating safety injection, a leak rate of 121.2 cfm would take over 48 minutes to reduce the RCS inventory by 5840 cu. ft.

The effect on core cooling during this time would be a reduction to the RCS cooldown rate.

A leak in the component cooling lines supplying the RHR heat exchangers, of the magnitude presented in 1.b above, would empty the CC surge tank in about 1 minute 43 seconds. Although air introduction into the CC System and cavitation of the CC pump would commence at this point, because of the large volume of water remaining in the CC return lines and still supplying the CC pump suction, a complete loss of CC flow to the RHR heat exchanger is not expected at this time.

The effect upon core cooling for this postulated event would be a reduction in the cooldown rate as the plant continues its cooldown on the remaining unaffected RHR/CC Systems.

3. Assuming the leak rate postulated in the response to 1.a above, a low pressurizer level alarm would be generated in about 41 seconds. A second lo-lo pressurizer level alarm would be generated less than 25 seconds after the first alarm. Additionally, a high sump level alarm in the affected vault would be actuated approximately 1 minute from the commencement of the leak.

   For the postulated leak in the component cooling lines identified in the 1.b response above, a component cooling surge tank low level alarm would activate in less than 1 minute. Within an additional 25 seconds, a lo-lo surge tank level alarm would also annunciate. Within 1-1/2 minutes from the onset of the leak, a high sump level alarm in the affected vault would also be actuated.

   For either of the above postulated leaks, the operator response would be to isolate the affected system and continue the RCS cooldown using the redundant, non-affected RHR and Component Cooling System.

440.45 The staff will require verification that no vortexing tendencies exist in the recirculation sump. Discuss the full scale preoperational tests which will show that under prototypical post-LOCA conditions, no adverse flow conditions will occur which could degrade ECCS pump performance. In lieu of full scale in plant tests, a scale model sump test may be acceptable to the staff. If you chose to conduct a scale model test, provide details of the test program. Include information of the model size, scaling principles utilized, comparison of model parameters to expected post-LOCA conditions, and a discussion on how all possible flow conditions and screen blockages will be considered in the model tests. The applicant should be advised that due to scaling problems, the staff will require that model test indicate considerable margin is available in respect to vortexing tendencies. Rotational flow patterns and surface dimples which might be acceptable in full scale tests, may not be acceptable in a model program.

RESPONSE: A scale model test program for the containment recirculation sumps was performed by the Alden Research Laboratories of the Worcester Polytechnic Institute. The reports of this test program have been submitted for NRC review. See the response to RAI #440.44.

In addition to the test program report identified above, the attached letter, Alden Research Laboratory to Yankee Atomic Electric Company, dated November 11, 1982, provides additional information relative to the acceptability of the Type I and II vortex formations noted during the model testing performed for Seabrook.

ALDEN RESEARCH LABORATORY
WORCESTER POLYTECHNIC INSTITUTE

November 11, 1982

Mr. Peter Anderson
Systems Engineering
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, MA 01709

REPLY TO NRC REVIEWER COMMENTS
SEABROOK/SUMP

Dear Mr. Anderson:

Regulatory Guide 1.82, dated June 1974 and titled "Sumps for Emergency Core Cooling and Containment Spray Systems," states "The sump design of the suction inlets should consider the avoidance of flow degradation by vortex formation." Vortex formation, per se, does not cause flow degradation; but rather, the ingestion of air by vortex action into the system is the cause of pump performance degradation (1, 2). Consequently, a containment sump hydraulic performance criterion regarding vortex formation in reduced scale model tests has been developed at the Alden Research Laboratory (ARL) to assure that air entrainment will not occur. This performance criterion has been that the maximum allowable vortex strength should be a coherent dye core vortex. It should be noted that a special test series on possible scale effects in modeling vortices was conducted as part of the recent parametric sump study for the NRC (9); and, this study showed no scale effects for large scale models of reactor sumps. However, to achieve an additional factor of safety, selected tests are conducted at flows higher than dictated by the relevant scaling laws.

The proposed revision of Regulatory Guide 1.82, which is dated February 1982 and was developed based partially on the generic containment sump studies done at the ARL (3), specifically states "Sump suction inlet design should consider the avoidance of air-ingesting vortices," in recognition of the cause of flow degradation. The sump hydraulic performance criterion used at the ARL eliminates the possibility of air ingesting vortices by limiting the acceptable vortex activity to a coherent dye core. This hydraulic performance criteria for containment sumps has been developed for and applied to several past studies which have been accepted by the NRC (4, 5, 6, 7, 8).

The ARL Classification Chart for Free Surface Vortex Activity, Figure 1, attempts to evaluate sump flow patterns over a wide range of conditions, and the defined flow patterns are not limited to actual vortex action. In the classification, vortex types 1 and 2 are not true vortices since they are limited to the surface motion only and do not interact with the suction inlet directly. The first true coherent vortex motion is the type 3, vortex which is a dye core occurring over the entire depth of the water column and usually entering the suction inlet.

In the Seabrook study, most vortex activity was limited to types 1 and 2, while the maximum vortex observed in only a few cases was a type 3 dye core vortex, which was noted to be "very intermittent and weak" or "surface activity only". In two previous studies (5, 7), a type 3 dye core vortex was noted in the final design. In both previous cases, as in the Seabrook study, the vortex was transient and occurred for only a few of the conditions tested. All other studies have had type 2 vortices. The NRC reviewers have not found the performance of any of the sumps studied to be unacceptable.

As mentioned above, the model performance criterion was selected conservatively, such that results from reduced scale models would provide adequate assurance of acceptable performance. For the Seabrook study, high temperature tests were conducted in conjunction with increased flows above Froude scaled flows, such that extrapolation to prototype Reynolds number at the correct Froude number could be accomplished. The only type 3, dye core vortices seen in the Seabrook study occurred for prototype velocity (twice the Froude scale velocity), indicating that the probability of a dye core vortex forming in the prototype was small. Also, since the coverplate over the sump was always submerged, any vortex action was outside the screens and was broken up by the screens.

The NRC-sponsored generic study of containment sumps (9) has likewise shown through tests with full scale and two reduced scale models, scale ratios of 1:2 and 1:4, that no scale effect exists for vortex formation and air ingestion for models operated at Froude scaled flows. Therefore, the Seabrook study results are a conservative indication of the expected vortex activity in the prototype.

The full scale parametric study on sump hydraulic performance will result in additional technical guidance for sump design based on a "maximum envelope" analysis (3), the data not considering a submerged solid sump cover plate. However, this guide is not intended to pre-empt testing a particular sump geometry to determine performance. In the case of the Seabrook sump, the geometry varies considerably from those tested in the generic study. The differences are: the sump coverplate for Seabrook was submerged, eliminating any free surface vortex action within the screens; and the vertical flow path for Seabrook is more complex due to the screen geometry.

A second concern about vortex formation may be the possibility of pump performance degradation by the creation of swirl, or rotating flow in the suction inlet. Direct measurements of this parameter were made in the model study and may be evaluated separately from the issue of surface vortex formation. In fact, several studies have shown that vortex formation and swirl in the suction inlet are not well correlated.

In summary, we wish to state that the Seabrook sump hydraulic characteristics are as favorable as any other final design model tested at the ARL.

Sincerely,

G. E. Hecker

George E. Hecker
Professor and Director

JBN/nm
Enclosures
cc: Mr. Henry Windgate

REFERENCES

1. Murakami, M., et al., "Flow of Entrained Air in Centrifugal Pumps," Proceedings of the 13th Congress of IAHR, Japan, August 31 - September 5, 1969, PPD8-1.

2. Patel, B.R., and Runstadler, P.W., "Investigations into Two-Phase Flow Behavior of Centrifugal Pumps," Proceedings, Symposium on Polyphase Flow in Turbomachinery, ASME Winter Annual Meeting, San Francisco, California, December 10-15, 1978.

3. Weigand, G.G., Krien, M.S., Webster, M.J., and Padmanabhan, M., "A Parametric Study of Containment Emerging Sump Performance," NUREG/CR-2758, SAND82-0624, ARL-46-82, July 1982.

4. Padmanabhan, M., "Hydraulic Model Studies of the Reactor Containment Building Sump, North Anna Nuclear Power Station - Unit 1," ARL Report No. 123-77/M250CR, July 1977.

5. Padmanabhan, M., "Assessment of Flow Characteristics Within a Reactor Containment Recirculation Sump Using a Scale Model," McGuire Nuclear Power Station, ARL Report No. 29-78/M208JF.

6. Padmanabhan, M., "Hydraulic Model Investigation of Vortexing and Swirl Within a Reactor Containment Recirculation Sump," Donald C. Cook Nuclear Power Station, ARL Report No. 108-78/M178FF.

7. Nystrom, J.B., "Experimental Evaluation of Flow Patterns in an RHR Sump with Simulation of Screen Blockage," ARL Report No. 24-81/M302LM.

8. Nystrom, J.B., "Model Study of Reactor Containment Sump Flow Characteristics, Virgil C. Sumner Nuclear Generating Station," ARL Report No. 47-81/M260EF.

9. Padmanabhan, M., and Hecker, G.E., "Assessment of Scale Effects on Vortexing, Swirl, and Inlet Losses in Large Scale Sump Models," NUREG/CR-2760, ARL 48-82, SAND82-7063, June 1982.

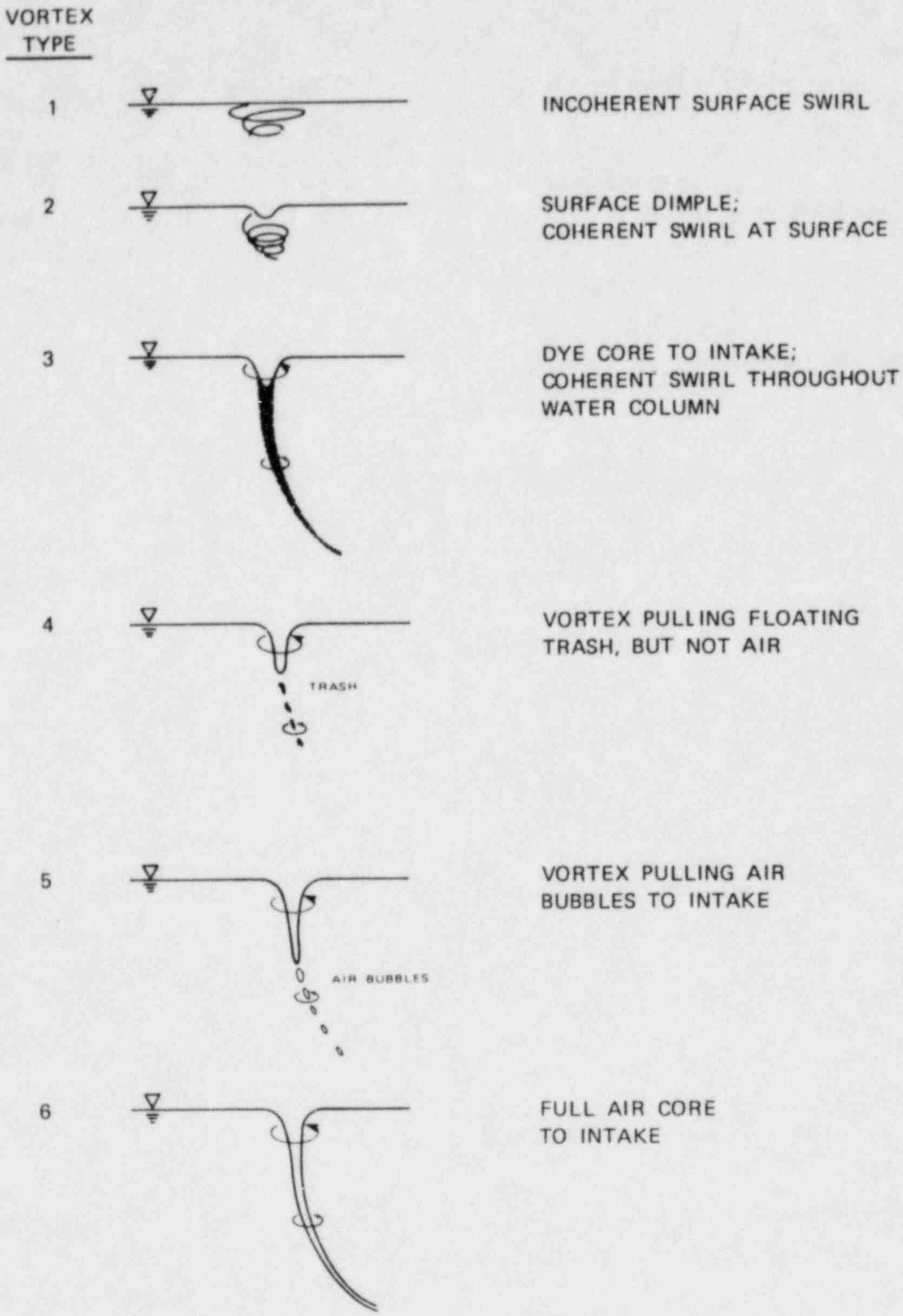

FIGURE 1 ARL VORTEX CLASSIFICATION CHART

440.52 During our review of license applications we have identified concerns related to the containment sump design and its effect on long-term cooling following a loss-of-coolant accident (LOCA).

These concerns are related to (1) creation of debris which could potentially block the sump screens and flow passages in the ECCS and the core; (2) inadequate NPSH of the pumps taking suction from the containment sump; (3) air entrainment from streams of water of steam which can cause loss of adequate NPSH; (4) formation of vortices which can cause loss of adequate NPSH, air entrainment and suction of floating debris into the ECCS; and (5) inadequate emergency procedures and operator training to enable a correct response to these problems. Preoperational recirculation tests performed by utilities have consistently identified the need for plant modifications.

The NRC has begun a generic program to resolve this issue. However, more immediate actions are required to assure greater reliability of safety system operation. We therefore require you take the following actions to provide additional assurance that long-term cooling of the reactor core can be achieved and maintained following a postulated LOCA.

1. Establish a procedure to perform an inspection of the containment, and the containment sump area in particular, to identify any materials which have the potential for becoming debris capable of blocking the containment sump when required for recirculation of coolant water. Typically, these materials consist of: plastic bags, step-off pads, health physics instrumentation, welding equipment, scaffolding, metal chips and screws, portable inspection lights, unsecured wood, construction materials and tools, as well as other miscellaneous loose equipment.

   "As Licensed" cleanliness should be assured prior to each startup.

   This inspection shall be performed at the end of each shutdown as soon as practical before containment isolation.

2. Institute an inspection program according to the requirements of Regulatory Guide 1.82, Item 14. This item addresses inspection of the containment sump components including screens and intake structures.

3. Develop and implement procedures for the operator which address both a possible vortexing problem (with consequent pump cavitation) and sump blockage due to debris. These procedures should address all likely scenarios and should list all instrumentation available to the operator (and its location) to aid in detecting problems which may arise, indications the operator should look for, and operator actions to mitigate these problems.

4. Pipe breaks, drain flow and channeling of spray flow released below or impinging on the containment water surface in the area of the sump can cause a variety of problems; for example, air entrainment, cavitation and vortex formation.

   Describe any changes you plan to make to reduce vortical flow in the neighborhood of the sump. Ideally, flow should approach uniformly from all directions.

5. Evaluate the extent to which the containment sump(s) in your plant meet the requirements for each of the items previously identified; namely debris, indadequate NPSH, air entrainment, vortex formation, and operator actions.

   The following additional guidance is provided for performing this evaluation:

   (1) Refer to the recommendations in Regulatory Guide 1.82 (Section C) which may be of assistance in performing this evaluation.

   (2) Provide a drawing showing the location of the drain sump relative to the containment sumps.

   (3) Provide the following information with your evaluation of debris:

      (a) Provide the size of openings in the fine screens and compare this with the minimum dimensions in the pumps which take suction from the sump (or torus), the minimum dimension in any spray nozzles and in the fuel assemblies in the reactor core or any other line in the recirculation flow path whose size is comparable to or smaller than the sump screen mesh size in order to show that no flow blockage will occur at any point past the screen.

      (b) Estimate the extent to which debris could block the trash rack or screens (50 percent limit). If a blockage problem is identified, describe the corrective actions you plan to take (replace insulation, enlarge cages, etc.).

      (c) For each type of thermal insulation used in the containment, provide the following information:

         (i) Type of material including composition and density,

         (ii) Manufacturer and brand name,

         (iii) Method of attachment,

(iv) Location and quantity in containment of each type,

(v) An estimate of the tendency of each type to form particles small enough to pass through the fine screen in the suction lines.

(d) Estimate what the effect of these insulation particles would be on the operability and performance of all pumps used for recirculation cooling. Address effects on pump seals and bearings.

RESPONSE: These concerns were presented as Enclosure 10 of Requests for Additional Information in a USNRC letter dated September 30, 1981, "Acceptance Review for Operating Licenses for Seabrook Station, Units 1 and 2", D. G. Eisenhut to W. C. Tallman. The response to Enclosure 10 was provided in Enclosure 3 to a PSNH letter dated November 27, 1981, "Response to Acceptance Review Requests for Additional Information (RAI's)" J. DeVincentis to D. G. Eisenhut.

Although the extensive sump model testing performed by Alden Laboratories confirmed that vortexing would not be a problem, even with up to 50% blockage of the sump screens due to debris, procedures will be developed which address both a possible vortexing problem and sump screen blockage. These procedures will address all likely scenarios, will list all instrumentation available to the operator (and their location) to aid in detecting problems which may arise, will provide indications which the operator should look for, and will provide recommended actions to mitigate these problems.

There are several lines in the west quadrant of the containment which are classed as high energy during normal plant operation:

1.	CS-328-2"	RC pump seal injection
2.	CS-329-2"	RC pump seal injection
3.	CS-360-4"	Letdown
4.	CS-355-3"	Charging
5.	NG-1652-1"	Accumulator nitrogen supply
6.	RC-13-12"	RHR pump suction
7.	RC-58-12"	RHR pump suction

All of these lines are, or can be, isolated if ruptured prior to the recirculation mode of post-accident operations.

There are also several lines which are classed as high energy following an accident (but not during normal plant operation):

1.	SI-272-3"	C.L. Injection from charging pump
2.	SI-273-1-1/2"	C.L. Injection from charging pump
3.	SI-251-4"	C.L. Injection from SI pump
4.	RH-155-8"	C.L. Injection from RHR pump
5.	RH-158-8"	C.L. Injection from RHR pump
6.	RH-160-8"	H.L. Injection from RHR pump

If a break occurs in one of these lines during recirculation mode, that train (or line) can be shutdown.

Since there are no high energy lines in the western quadrant which cannot be isolated, the potential for vortexing air entrainment due to a pipe break is negligible.

The effect of the volume of water entrapped in the containment which would otherwise contribute to NPSH available to the ESF pumps has been factored into the NPSH calculation for the pumps, as described in Section 6.2.1.1.(b).6. In addition to the entrapped water, there are drain lines equipped with strainers (also described in this section) which permit a flow path between the reactor cavity and refueling canals to elevations above the water level in the rest of the containment. Should the strainers on these lines become blocked, an additional volume of 5760 cubic feet of water would be trapped. The resulting reduction of water height would be 5.76 inches. This height reduction has not been factored into the NPSH available calculation as presented in Sections 6.2.2.2.g and j for the CBS pumps and Section 6.3.2.2.d for the RHR pumps. However, incorporation of this height reduction still results in the available NPSH being greater than the required NPSH at maximum design runout flow conditions (see the above referenced sections and revised RAI 440.39 response forwarded by PSNH letter dated November 8, 1982, "Revised Responses to 440 Series RAI's Reactor Systems Branch", J. DeVincentis to G. W. Knighton).

The effect of the volume of water entrapped in the containment on decay heat removal capability is limited to potential peak sump water temperature effects since adequate NPSH as discussed above results in adequate ESF pump flowrates for cooldown following the accident. The sump water peak temperature analyses as illustrated for various accident scenarios on Figures 6.2-3, 6.2-6, 6.2-9, 6.2-12, 6.2-15 and 6.2-18 include the potential entrapped water volumes as part of the recirculated inventory immediately upon initiation of the recirculation mode. In actuality, the entrapped water would not immediately enter the recirculated water inventory, but would eventually mix with this inventory because spray and water flow from the break would displace the entrapped water by overflow as recirculation continues. Since initial entrapped water would be high temperature LOCA fluid and the mixing of this water with the cooler recirculated water is delayed, the highest sump temperature during the recirculation mode would occur at a later time than calculated, thereby resulting in lower peak temperatures during this mode because of the further progression of the cooldown.