TVA

Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

December 21, 2007

TVA-SQN-TS-07-04 10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Gentlemen:

In the Matter of)
Tennessee Valley Authority (TVA)) Docket No. 50-328

SEQUOYAH NUCLEAR PLANT (SQN) - UNIT 2 - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING LARGE BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODS (TAC NO. MD6259)

References: (1) TVA letter to NRC dated July 26, 2007, "Sequoyah Nuclear Plant (SQN) - Unit 2 - Technical Specifications (TS) Change 07-04 'Revision of Core Operating Limits Report (COLR) References for Realistic Large Break Loss of Coolant Accident Methodology'"

(2) TVA letter to NRC dated October 3, 2007, "Sequoyah Nuclear Plant (SQN) - Unit 2 - Technical Specifications (TS) Change 07-04 'Revision of Core Operating Limits Report (COLR) References for Realistic Large Break Loss of Coolant Accident Methodology Supplemental Information'"

This letter responds to the NRC's request for additional information dated November 30, 2007, associated with the TS change request in the referenced letters. The enclosure provides TVA's responses to NRC's request. There are no commitments contained in this letter.

Please note that a number of the enclosed responses discuss changes that will be made to the S-RELAP5 computer code used in the SQN Unit 2 realistic large break loss-of-coolant accident (RLBLOCA) analysis to address generic NRC issues

A001
NRR

with the code (i.e., rod quench limits, Forslund-Rohsenow heat transfer coefficient limits and break size limits). To provide confirmation of the enclosed responses for these issues for the SQN Unit 2 application, the current RLBLOCA analysis documented in Topical Report No. ANP-2655(P) Revision 00 will be ran with the updated S-RELAP5 computer code.

The re-run with the updated computer code will be performed using 2 sets of 59 cases. One set will assume the availability of off-site power and one set will assume off-site power is not available consistent with the supplemental information submitted by Reference 2. The results of the SQN Unit 2 application re-run will be presented in a formal revision to Topical Report No. ANP-2655(P), which is scheduled to be submitted to NRC in February 2008.

If you have any questions about this change, please contact me at 843-7170.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 21st day of December, 2007.

Sincerely,

James D. Smith
Manager, Site Licensing and
Industry Affairs

Enclosure
cc (Enclosure):

Mr. Brendan T. Moroney, Senior Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop 08G-9a
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852-2739

Mr. Lawrence E. Nanney, Director
Division of Radiological Health
Third Floor
L&C Annex
401 Church Street
Nashville, Tennessee 37243-1532

ENCLOSURE

TENNESSEE VALLEY AUTHORITY (TVA)
SEQUOYAH NUCLEAR PLANT (SQN)
UNIT 2

NRC Request:

1. Reactor Power - Table 3.3, Item 2.1, and its associated Footnote 1 indicate that the assumed reactor core power "includes uncertainties." The use of a reactor power assumption other than 102 percent, regardless of BE or Appendix K methodology, is permitted by Title 10 of the *Code of Federal Regulations (10* CFR), Part 50, Appendix K.I.A, "Required and Acceptable Features of The Evaluation Models, 'Sources of Heat During a LOCA." However, Appendix K.I.A also states: "... An assumed power level lower than the level specified in this paragraph [1.02 times the licensed power level], **(but not less than the licensed power level)** may be used provided..."

   Please clarify what is meant by "includes uncertainties." What are the uncertainties, and how are the uncertainties included?

TVA Response

As indicated in Item 2.1 of Table 3.3 of ANP-2655(P) Revision 00, the assumed reactor core power for the SQN realistic large break loss-of-coolant accident is 3479 MWt. This value represents the plant rated thermal power (i.e., total reactor core heat transfer rate to the reactor coolant system) of 3455 MWt with a maximum power measurement uncertainty of 0.7 percent (24 MWt) added to the rated thermal power.

The power measurement uncertainty assumption discussed in 10CFR50, Appendix K was previously reduced for SQN from 2.0 percent of the plant rated thermal power to 0.7 percent based on the installation of a leading edge flow meter (LEFM) system to measure main feedwater flow. The improved feedwater flow measurement accuracy provided by the LEFM allowed for a power measurement uncertainty recovery of 1.3 percent. The basis for the current 0.7 percent measurement uncertainty assumption is documented in Topical Report No. WCAP-15669, Revision 00. This report was submitted to NRC as part of SQN Technical Specification Change Request No. TVA-SQN-TS-01-08 by a TVA letter dated November 15, 2001. NRC review and acceptance of the current power measurement uncertainty is documented in the NRC Safety Evaluation Report (SER) issued for Technical Specification Change Request No. TVA-SQN-TS-01-08 dated April 30, 2002.

NRC Request

2. Does the version of SRELAP used to perform the computer runs assure that the void fraction is less than 95 percent, and the fuel cladding temperature is less than 900 degrees Fahrenheit (°F) before it allows rod quench?

TVA Response
No, the version of S-RELAP employed for the SQN Unit 2 License Amendment Request (LAR) only restricts quenching to cladding temperatures below Tmin. There is no restriction on local

void fraction. Peak cladding temperatures and significant cladding oxidation occur at void fractions above 98 percent. Cladding quench occurs after a substantial cladding cooldown accompanied by significant decreases in local void fractions. Because of the timing and prototypical void fractions at which quench occurs during a reflooding or refilling condition, it was not considered necessary to limit the quench process by local void fraction.

Quench can be, however, a chaotic process that covers a span of time during which the cladding migrates in and out of the transition to nucleate boiling. Thus, particularly for low pressure situations, it is more appropriate to describe an average void fraction during the process of quench than an instantaneous value. Table 2-1 provides the Tmin for each of the top 10 highest PCT cases from the SQN Unit 2 RLBLOCA case set along with an instantaneous void fraction at the start of the quench process and the approximate average local void fraction during the quench process for the PCT location. For 5 of the 10 cases, the instantaneous void fraction at the time of quench approaches or exceeds 0.95. However, the average voiding through the quench process is uniformly below 0.95. For the revised RLBLOCA analysis discussed in the submittal letter, a coding change in S-RELAP5 will be implemented that will not allow quench to proceed for void fractions above 0.95.

Table 2-1: T_{min} and Quench Void Fraction for Upper Ten PCT Cases

PCT Ranking	Case Set Number	T_{min} °F	Void Fraction at the Start of Quench	Approximate Average Void Fraction During Quench
1	44	623	0.92	0.80
2	23	604	0.80	0.85
3	3	657	0.97	0.85
4	55	702	0.95	0.85
5	43	646	0.94	0.85
6	6	666	0.83	0.70
7	45	655	0.92	0.75
8	26	632	0.87	0.77
9	22	714	0.95	0.87
10	8	653	0.97	0.85

NRC Request

3. Provide justification that the SRELAP rod-to-rod thermal radiation model applies to the SQN-2 core.

TVA Response

The EMF-2103(P)(A) Realistic LBLOCA methodology does not provide modeling of rod-to-rod radiation. The fuel rod surface heat transfer processes included in the solution at high temperatures are: film boiling, convection to steam, rod to liquid radiation and rod to vapor radiation. This heat transfer package was benchmarked against various experimental data sets involving both moderate (1600 °F – 2000 °F) and high (2000 °F to over 2200 °F) peak cladding temperatures and shown to be conservative when applied nominally. The normal distribution of the experimental data was then determined. During the execution of a RLBLOCA evaluation, the heat transferred from a fuel rod is determined by the application of a multiplier to the nominal heat transfer model. This multiplier is determined by a random sampling of the normal distribution of the experimental data benchmarked. Because the data benchmarked includes the effects of rod-to-rod radiation, and the nominal heat transfer modeling is conservative relative to the benchmarked data, it is reasonable to conclude that the modeling implicitly includes a conservative allocation for rod-to-rod effects.

Notwithstanding any conservatism evidenced by experimental benchmarks, the application of the model to commercial nuclear power plants provides some additional margins due to limitations within the experiments. The benchmarked experiments, FLECHET SEASET and ORNL THTF, used to assess the S-RELAP heat transfer model incorporated constant rod powers across the experimental assembly. Temperature differences that occurred were the result of guide tube, shroud or local heat transfer effects. In the operation of a pressurized water reactor (PWR) and in the RLBLOCA evaluation, a radial local peaking factor is present, creating power differences that tend to enhance the temperature differences between rods. In turn, these temperature differences lead to increases in net radiation heat transfer from the hotter rods. The expected rod-to-rod radiation will likely exceed that embodied within the experimental results. Therefore, the implicit application of rod-to-rod radiation of the EMF-2103(P)(A) RLBLOCA methodology is more conservative.

In summary, the conservatism of the heat transfer modeling established by benchmark can be reasonably extended to plant applications, and the plant local peaking provides a physical reason why rod-to-rod radiation should be more substantial within a plant environment than in the test environment. Therefore, the lack of an explicit rod-to-rod radiation model, in the version of S-RELAP applied to SQN Unit 2, does not invalidate the conclusion that the cladding temperature and local cladding oxidation have been demonstrated to meet the criteria of 10 CFR 50.46 with a high level of probability.

NRC Request

4. In the SQN-2 calculations, is the Forslund-Rohsenow model contribution to the heat transfer coefficient limited to less than or equal to 15 percent when the void fraction is greater than or equal to 0.9?

TVA Response

EMF-2103(P)(A), Revision 0, does not require a limitation on the Forslund-Rohsenow heat transfer correlation. However, as a practical matter, the contribution of Forslund-Rohsenow to the total heat transfer package decreases as the local void fraction increases and is less than 15 percent for void fractions above approximately 97 percent. Figure 4-1 provides a scatter plot, taken from the limiting SQN Unit 2 case, of a sampling of the percentage of the Forslund-Rohsenow contribution to heat transfer as a function of liquid fraction. The sampling includes several points near the time of PCT. As expected, the fluid cooling the hot spot at the time of PCT comprises steam in transition to low liquid fraction. The contribution of Forslund-Rohsenow to the total heat transfer in such a flow is limited, and well below 15 percent. Thus, it can be concluded that there would be no direct influence of a limitation on Forslund-Rohsenow on the resultant limiting PCT for SQN Unit 2.

An indirect effect on PCT may occur because such a limitation may reduce steam generation below the hot spot where the void fraction could reside between 90 percent and 97 percent. Such an impact is expected to be small and does not pose a concern for the conclusions reached by the RLBLOCA evaluation.

In summary, there is no limitation of the contribution of Forslund-Rohsenow to the heat transfer for the SQN Unit 2 RLBLOCA analysis. However, the impact of such a limitation, if it were to be applied, would be minimal and would not lead to an alteration of the conclusions of the analysis that the cladding temperature and local cladding oxidation have been demonstrated to meet the criteria of 10 CFR 50.46 with a high level of probability. For the revised RLBLOCA analysis discussed in the submittal letter, a coding change in S-RELAP5 will be implemented that will not allow the contribution of Forslund-Rohsenow to exceed 15 percent of the total heat transfer.

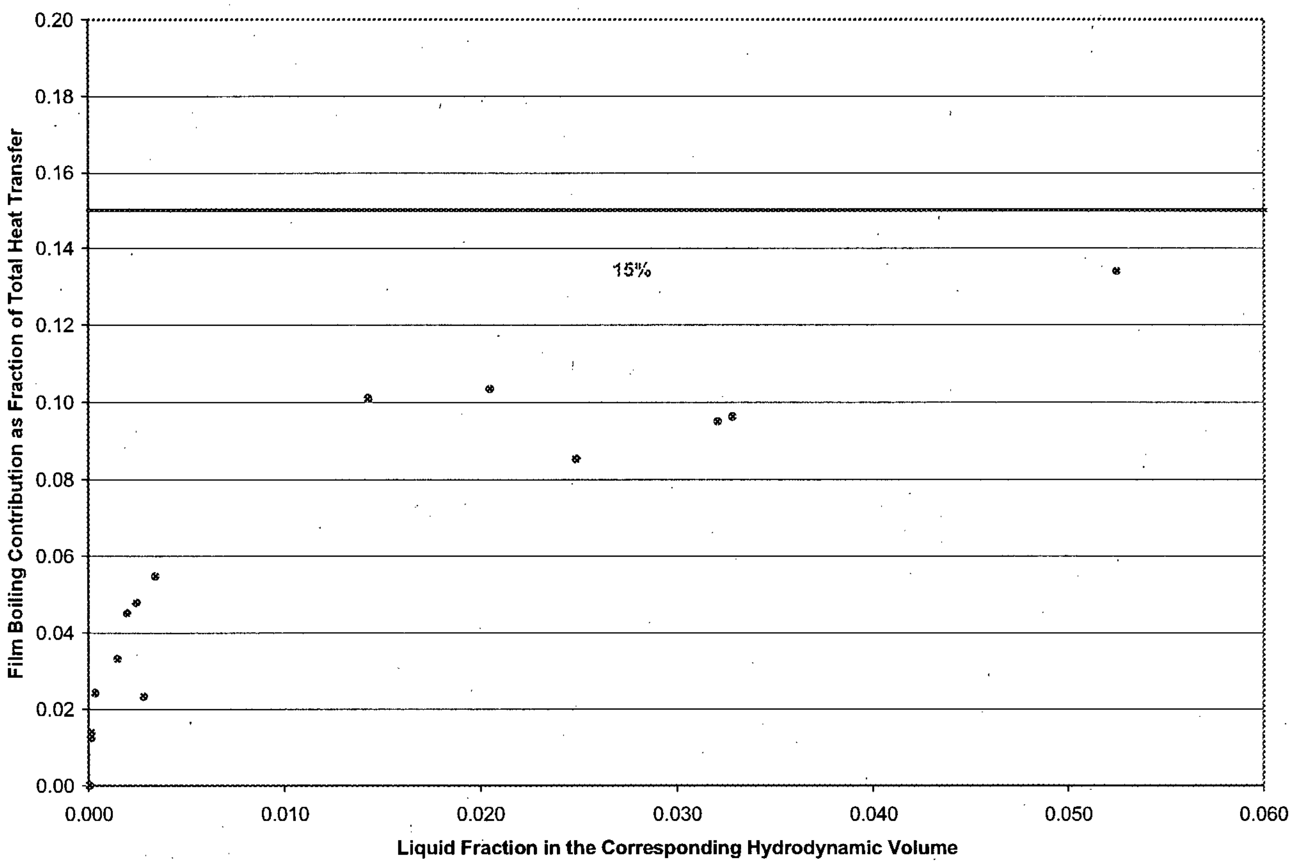

Figure 4-1: Film Boiling Heat Transfer Contribution – Forslund-Rohsenow Limit

PCT Node 35 @~PCT Time (99s) ± 5 Seconds

NRC Request

5. Was the downcomer model for the SQN-2 design rebenchmarked, performing sensitivity studies, assuming adequate nodding in the downcomer, the water volume, the vessel wall, and other heat structures, with all heat structures' initial temperature at or greater than 1800 °F, and containment pressure less than 90 pounds per square inch absolute?

TVA Response

The downcomer model for SQN Unit 2 has been established generically as adequate for the computation of downcomer phenomena including the prediction of potential local boiling effects. The model was benchmarked against the UPTF tests and the LOFT facility in EMF-2103(A)(P) Revision 0. Furthermore, AREVA addressed the effects of boiling in the downcomer in a letter from Jim Malay to NRC dated April 4, 2003. The letter cites the lack of direct experimental evidence but contains sensitivity studies on high and low pressure containments, the impact of additional azimuthal noding within the downcomer, and the influence of flow loss coefficients. Of these, the study on azimuthal is most germane to this question; indicating that additional azimuthal nodalization allows higher liquid buildup in portions of the downcomer away from the broken cold leg and increases the liquid driving head.

AREVA is currently conducting downcomer axial noding and wall heat release studies. The heat release studies comprise an S-RELAP5 sensitivity with the wall heat structure mesh spacing cut in half and a comparison to a closed form solution for wall heat release. The preliminary results confirm the applicability of the Revision 0 modeling. However, the work is on going and not yet suitable for documentation herein. The results will be provided along with the revised RLBLOCA analysis discussed in the submittal letter, to be submitted in February, 2008. The adequacy of the Revision 0 downcomer noding discussed in the April 4, 2003, Malay to NRC letter will be further verified in the submittal of the revised RLBLOCA analysis.

NRC Request

6. Were all break sizes assumed greater than or equal to 1.0 square foot?

TVA Response

No, EMF-2103(P)(A), Revision 0, sets the minimum total break size at 0.1 times A_{pipe}, 0.41 ft^2, for SQN Unit 2. However, the SQN Unit 2 case set contains only one case with a total break area below 1.0 ft^2. The total break area for that case was 0.81 ft^2 and the PCT was 662°F. For the revised RLBLOCA analysis discussed in the submittal letter, a change to the range over which break areas are sampled will not allow break areas below 1.0 ft^2.

Note: According to EMF-2103(P)(A), Revision 0, methodology, the break area (guillotine or split type break) is referenced as one half of the total opening between the reactor coolant system and the containment. Thus, Figure 3-7 and Figure 3-9 within ANP-2655(P) that refer to break area will show several breaks at less than 1.0 ft^2 and one at less than 0.5 ft^2. These values represent half of the total break area. The one case at less than 0.5 ft^2 is the case referenced in the above response with a total break area of 0.81 ft^2.

NRC Request

7. Verify that the SQN-2 ICECON model is that shown in Figure 5.1 of EMF-CC-39(P) Revision 2, "ICECON: A Computer Program Used to Calculate Containment Back Pressure for LOCA Analysis (Including Ice Condenser Plants)."

TVA Response

Yes, the ICECON model used for the SQN Unit 2 RLBLOCA evaluation is that shown in Figure 5.1 of EMF-CC-39 (P). Two model features present in the figure, the lower compartment spray flow and the upper compartment to lower compartment atmosphere flow are either not present in SQN Unit 2 or not activated prior to the end of the RLBLOCA analysis and are, therefore, not activated in the SQN Unit 2 RLBLOCA containment model.

NRC Request

8. In order to conduct its review of the SQN- 2 application of AREVA's realistic LBLOCA methods in an efficient manner, the NRC staff would like to make reference to the responses to the NRC staff's requests for additional information that were developed for the application of the AREVA methods to the North Anna Power Station, Unit Nos. 1 and 2, and found acceptable during that review (the NRC Staff safety evaluation was published on April 1, 2004). The staff would like to make use of the information that was provided by the North Anna licensee that is not applicable only to North Anna or only to subatmospheric containments. This information is contained in letters to the NRC from the North Anna licensee (Docket numbers 50-338 and 50- 339) dated September 26, 2003 and November 10, 2003. The specific responses that the staff would like to reference are:

September 26, 2003 letter:	NRC Question 1
	NRC Question 2
	NRC Question 4
	NRC Question 6
November 10, 2003 letter:	NRC Question 1

Please verify that the information in these letters is applicable to the AREVA model applied to SQN-2, except for that information related specifically to North Anna, and to sub-atmospheric containments.

TVA Response

The responses provided in ML032790396 questions 1, 2, 4, and 6 are for the most part generic and related to the ability of ICECON to calculate containment pressures. Excepting as follows, they are applicable to the SQN Unit 2 RLBLOCA submittal.

Question 1 – Completely Applicable
Question 2 – Completely Applicable
Question 4 – Completely Applicable (the reference to CSB 6-1 should now be to CSB Technical Position 6-2). The NRC altered the identification of this branch technical position in Revision 3 of NuReg-0800).
Question 6 - The direct response is completely applicable excepting that the reference to "North Anna Units 1 and 2" should be deleted. The statement in which the North Anna units are referenced is equally valid without identification of any specific plant.

The supplemental request and response are specific to North Anna and are not applicable to SQN Unit 2.

The response provided in ML033240451 question 1 contains both generic and plant specific content. The portions that are generic remain applicable to SQN Unit 2. However, the North Anna Units use sub-atmospheric containment designs and SQN Unit 2 is of the ice condenser type. This leads to several differences in the way the information would be presented. If the NRC requires the level of detail provided in this response for SQN Unit 2, such a response can be generated and submitted along with the revised submittal in February.

NRC Request

9. ANP-2655(P) shows that the containment parameters treated statistically are: (1) upper compartment containment volume, (2) upper compartment containment temperature, and (3) lower compartment containment temperature. ANP-2655(P) states that "in many instances" the guidance of NRC Branch Technical Position CSB 6-1 was used in determining the other containment parameters.

   (a) How is the mixing of containment steam and ice melt modeled so as to minimize the containment pressure?

   (b) Verify that all containment spray and fan coolers are assumed operating at maximum heat removal capacity.

   (c) Describe how the limits on the volume of the upper containment are determined.

   (d) How are the containment air return fans modeled; and what is the effect of this modeling on the containment pressure?

   (e) Describe how passive heat sink areas and heat capacities are modeled so as to minimize containment pressure.

   (f) What value is used for the initial ice mass? Does this value result in minimizing containment pressure?

TVA Response

(a) In the ice chest, the water formed by melted ice and condensed steam flows to the lower ice chest plenum where it accumulates if the ice bay drains are not large enough to accommodate the rate of water production, Figure 9-1. When the water level in the lower ice chest plenum rises above the bottom of the lower doors, water spillage through the lower doors as well as through the drain ports occurs. The water drainage (spillage plus drainage) from the ice chest falls through the lower compartment vapor. This condenses steam and reduces the containment pressure. The ice chest drainage flow is treated as 100 percent efficient spray during the post-blowdown period of the transient.

(b) SQN Unit 2 does not include fan coolers in its building pressure control. The building spray system is modeled at maximum heat removal capacity. Both systems are activated at 100 percent capacity (7700 gpm) within 10 seconds of the building pressure signal. The spray temperature is 55 °F (5 °F below the minimum allowed by technical specifications) and the spray efficiency is 100 percent.

(c) Te upper containment volume is sampled from 651,000 ft^3 to 692,600 ft^3. The minimum value is carried over from use in the long term containment integrity analysis of record for SQN. The maximum value is computed as the volume available within the upper dome of the containment and within the crane wall above the control rod drive missile shield with no internal structures accounted. Figure 9-2 provides an illustration of these volumes within the overall containment.

(d) Because the start time for the recirculation fan is 600 seconds and all of the RLBLOCA cases quench by 500 seconds, the air flow from the upper compartment to the lower compartment is not modeled (no flow allowed). This approach is conservative in that no bypass of the ice beds (from lower to upper compartments) is allowed in the short term. All flow between lower and upper compartments is directed through the ice beds.

(e) The passive heat sinks were derived from a SQN Unit 2 minimum containment pressure model approved for use with deterministic, Appendix K, LOCA evaluations. The surface area of these heat structures was increased by 5 percent to provide a further margin toward lower pressures. Surface coatings, where they existed, were modeled as the underlying material so that the insolating effect on a metallic structure was discounted. As a further pressure lowering measure, condensing heat transfer was modeled on those structures which fall below the transient water level during LOCA.

(f) The initial ice mass was 2.448 million pounds. This mass is characterized as a nominal value (10 percent more than the minimum of ~2.226 million pounds) and is not necessarily bounding of the possible mass in the ice chests. However, less than half of the ice (~1.2 million pounds) is calculated to melt during the LOCA transients (i.e., first 500 seconds of the transient). Therefore, the ice mass used in the ICECON simulation is adequate for the prediction of containment pressures (break back pressures) during LOCA.

NRC Request

10. Provide a statement confirming that TVA and its LBLOCA analyses vendor have ongoing processes that assure that the input variables and ranges of parameters for the SQN-2 LBLOCA analyses conservatively bound the values and ranges of those parameters for the as operated SQN-2 plant. This statement addresses certain programmatic requirements of 10 CFR 50.46, Section (c).

TVA Response

TVA and the LBLOCA Analysis Vendor have an ongoing process to ensure that all input variables and parameter ranges for the SQN Unit 2 realistic large break loss-of-coolant accident are verified as conservative with respect to plant operating and design conditions. In accordance with TVA Quality Assurance program requirements, this process involves 1) definition of the required input variables and parameter ranges by the Analysis Vendor, 2) compilation of the specific values from existing plant design input and output documents by TVA and Vendor personnel in a formal analysis input summary document issued by the Analysis Vendor, and 3) formal review and approval of the input summary document by TVA. Formal TVA approval of the input document serves as the release for the Vendor to perform the analysis.

Continuing review of the input summary document is performed by TVA as part of the plant design change process and cycle specific core design process. Changes to the input summary required to support plant modifications or cycle specific core alternations are formally communicated to the Analysis Vendor by TVA. Revisions and updates to the analysis parameters are documented and approved in accordance with the process described above for the initial analysis.

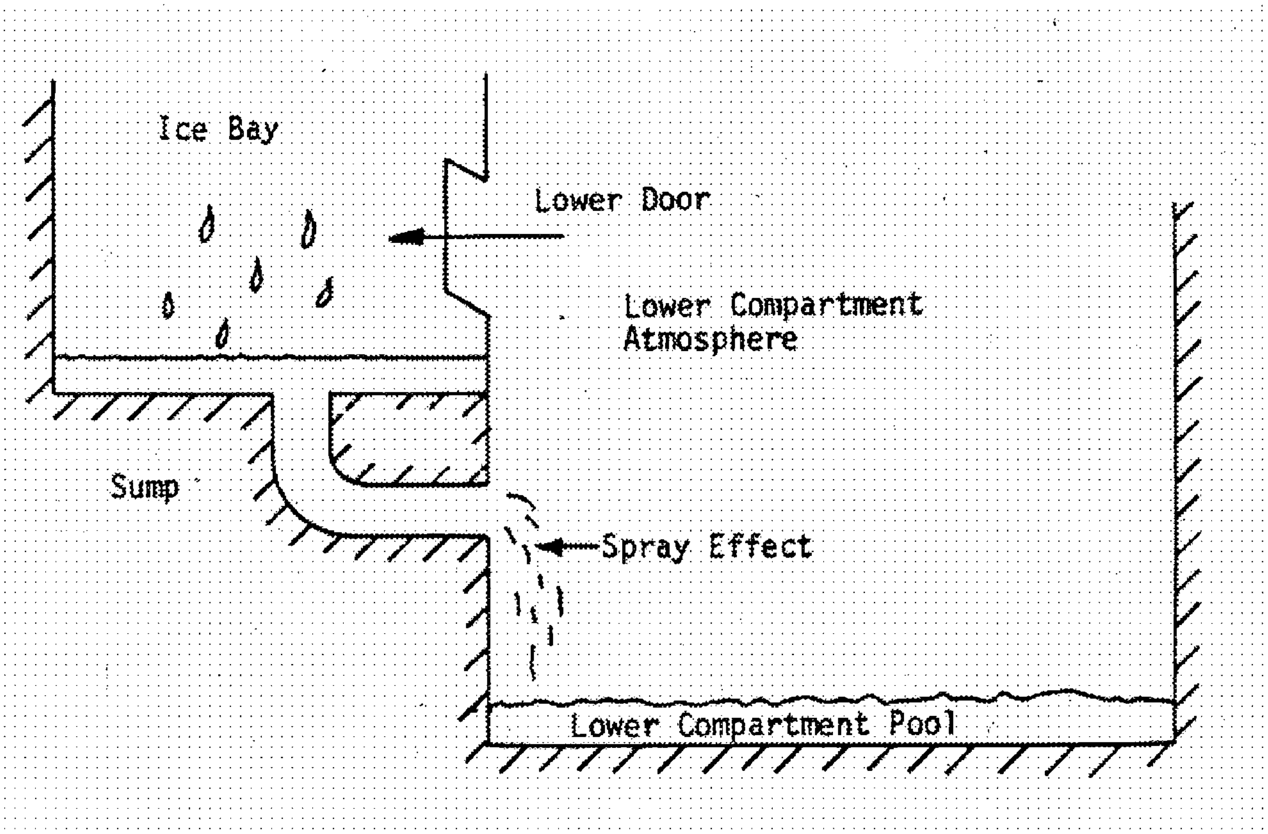

Figure 9-1: Ice Condenser Sump Configuration

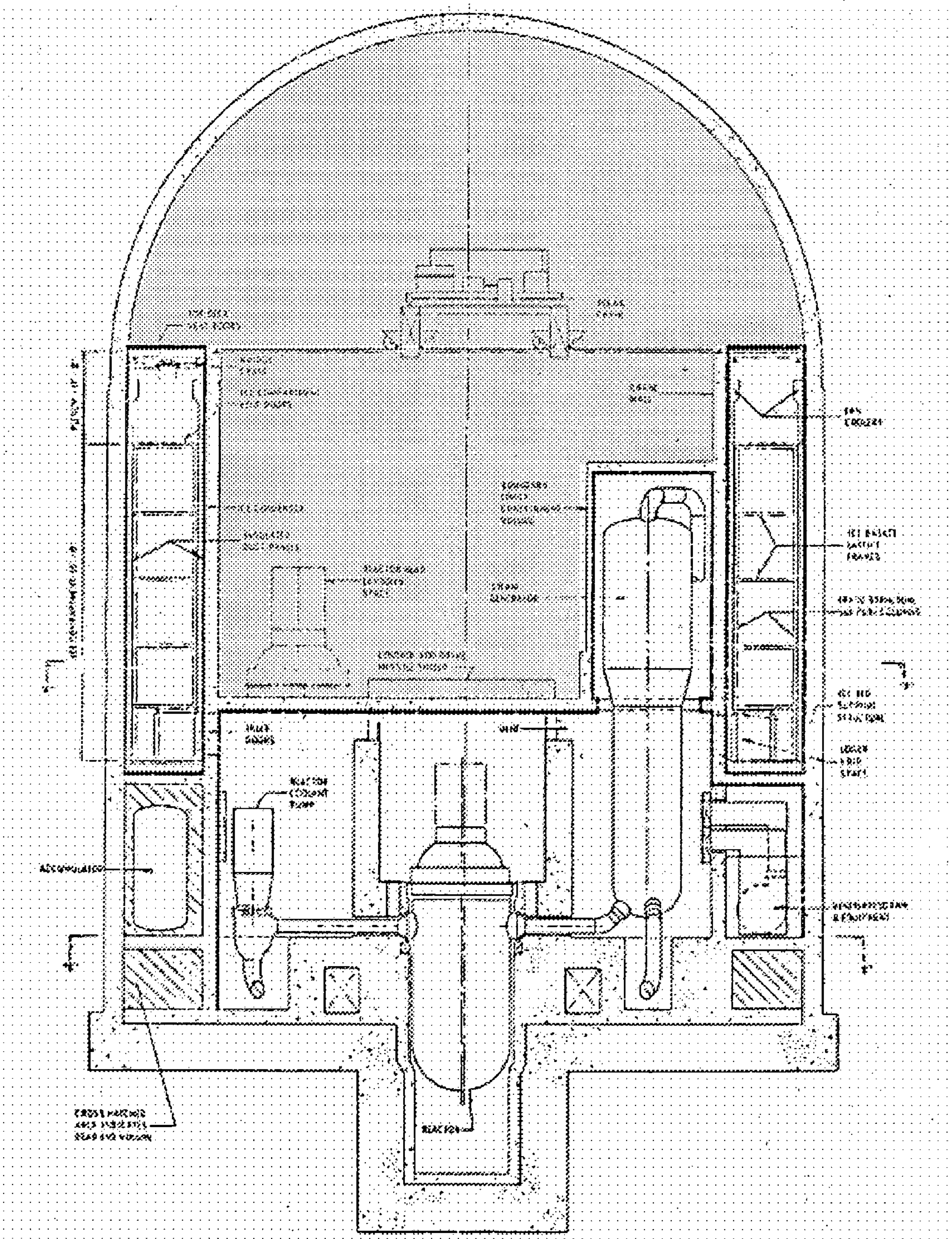

Figure 9-2: Representative Containment Arrangement for Ice Condenser Plants