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MAY 05 1994

Director of Nuclear Reactor Regulation
Attn: Mr. C. L. Miller, Project Director
Project Directorate I-2
Division of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
CONCERNING LOSS OF SPENT FUEL POOL COOLING
PLA-4134 FILE R41-2

Docket Nos. 50-387 and 50-388

Reference: *NRC Letter, J. W. Shea to R. G. Byram "Request for Additional Information (RAI) concerning Loss of Spent Fuel Pool Cooling, Susquehanna Steam Electric Station, Units 1 and 2 (TAC NO. M85337)," dated April 21, 1994.*

Dear Mr. Miller:

In the above reference, the staff determined that the addition of make-up water to the spent fuel pool (SFP) from the emergency service water (ESW) system during a design basis loss of coolant accident (LOCA) and/or loss-of-offsite power (LOOP) event is a commitment documented in the licensing basis for the Susquehanna SES (SSES). Based on this determination, the staff requested information to demonstrate that this make-up can be supplied to the spent fuel pools from the ESW system under design basis accident (DBA) conditions (specifically, design basis LOCA).

As stated in previous PP&L submittals, the engineered safety grade make-up system for fuel pool cooling for Susquehanna SES is the Emergency Service Water system. ESW is a system common to both Units with valves, provided to direct fuel pool make-up, located in each unit. The use of ESW for supplying fuel pool make-up involves the manipulation of three manual valves (2") per loop of ESW. Susquehanna SES also has a common refueling floor with the spent fuel pools of both units hydraulically connected so that water from one unit's pool can flow to the other unit.

Initiation of ESW make-up to the non-accident SFP will result in both pools being filled when water level is raised above the height of the weirs in the spent fuel pool for the non-accident unit. With the pools isolated, water added to one pool will overflow to its skimmer surge tank; which when completely filled will overflow to the cask storage pit; which will overflow to the accident unit's skimmer surge tank; which in turn will overflow into the accident unit's SFP. Instructions exist in the procedures to fill above the weirs, thereby completely filling the skimmer surge tank.

In addition to ESW, there are several systems which can be used to provide water into the fuel pool based on system design and/or location. Specifically, make-up water can be added from the Condensate Storage Tank (CST) or Residual Heat Removal Service Water (RHRSW). Demineralized Water and Fire Water can also be called upon to add water as required. Thus, fuel pool make-up for both units can be accommodated in either Unit. In response to a LOCA event in one Unit, we would provide fuel pool make-up from the non-accident Unit and not require operators to be exposed to accident conditions.

However, PP&L has analyzed access to the ESW valves in the accident unit to address the unlikely situation that the non-accident unit ESW valves are unavailable and has concluded that even using a Regulatory Guide 1.3 source term, operator access is possible. Specific details are provided in the attached report. This report, as previously stated, also addresses all seven (7) NRC's questions contained in the RAI, question by question. It should be noted that the specific scenario is not the same for each question. This is due to the nature of the questions asked within the RAI, which requested specific information for a seismic event and LOCA events both within and outside the SSES licensing basis. Therefore, the information provided as a response to one question should not be automatically applied to the next.

The actions required to support use of ESW in response to a loss of SFP cooling are identified in plant procedures generated to support operation of both SSES Units. Procedures enhancements have been made to provide greater attention to a loss of SFP cooling and address the need to restore cooling. These procedure enhancements are also incorporated into the operator re-qualification training program at SSES.

Therefore, PP&L concludes that :

- With respect to makeup for a DBA/LOCA Susquehanna SES has multiple means to assure make-up.
- The Emergency Service Water valves that support make-up to the Spent Fuel Pool have been determined to be accessible from the affected unit (i.e. <5 Rem) even if a Regulatory Guide 1.3 source term is assumed.
- Emergency Service Water make-up to one Spent Fuel Pool will eventually fill both pools. Therefore, access to the accident unit is not required to provide make-up to its Spent Fuel Pool.

Questions regarding this response should be directed to Mr. J. M. Kenny at (610) 774-7904.

Very truly yours,

R. G. Byram

Attachment

cc: NRC Document Control Desk (original)
NRC Region I
Mr. G. S. Barber, NRC Sr. Resident Inspector
Mr. C. Poslusny, Jr., NRC Project Manager

ATTACHMENT to PLA-4134

RESPONSE TO NRC APRIL 21, 1994 RAI CONCERNING LOSS OF SFP COOLING

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 AND 2

May 5, 1994

ATTACHMENT to PLA-4134

This document provides PP&L's response to the seven (7) specific questions identified in the enclosure to the NRC's April 21, 1994 Request for Additional Information (RAI) concerning a Loss of Spent Fuel Pool Cooling. It should be noted that the specific scenario is not the same for each question. This is due to the nature of the questions asked by the NRC, which requested specific information for both seismic and LOCA events. Therefore, the information provided in response to one question should not be applied to another question.

NRC QUESTION 1

In a letter dated May 24, 1993, PP&L stated that "the normal SFP cooling system will automatically be shed from the plant electrical system, along with other non-safety-related equipment, to permit the start-up of the large emergency core cooling system (ECCS) pumps on the LOCA unit." In a letter dated March 21, 1994, PP&L provided clarification that indicated that the service water pumps are lost due to the auxiliary load shed feature and that the cooling function of the fuel pool cooling system will no longer be provided for the accident unit only.

The staff requests that you provide a comprehensive description of the response and operation of the spent fuel pool cooling and cleanup system following a LOCA. Please address the following issues as a minimum:

a. The March 21, 1994 letter indicated that loads such as the service water system can be restored 10 minutes after initiation of the event and that complete restoration of the service water system would take approximately one 12-hour shift. Please describe what activities, including relevant procedures, would be specifically required to restore service water to the fuel pool cooling system and the expected duration of these activities.

b. Describe the expected response and operation of the fuel pool cooling system following the loss of service water. Describe the expected system response, including system heat-up and any impact of system heat-up as well as expected or necessary operator manipulation of the fuel pool cooling system following the loss and restoration of service water.

RESPONSE TO NRC QUESTION 1a

This response assumes a LOCA occurs while the plant is operating at 100% steady state power. A Loss of Offsite Power (LOOP) is <u>not</u> assumed to occur coincident. As a result of the LOCA, a generator lockout will occur resulting in an auxiliary equipment load shed ("aux load shed"). The "aux load shed" will result in the trip of the Service Water (SW) pump supply breakers, thereby terminating cooling water flow to the SFP cooling heat exchangers. The Service Water pumps are supplied power from the 13.8 KV switchgear 1A101 and 1A102 which is located on the 699' elevation of the turbine building. The pump supply breaker lock out relays have to be manually reset (they do not automatically reset). The supply breakers for the Spent Fuel Pool Cooling (SFPC) pumps will <u>not</u> be tripped by the aux load shed and will continue to operate even though it will no longer be providing cooling to the LOCA Unit's SFP. The non-LOCA Unit's Service Water (SW) and SFPC systems will be unaffected by the LOCA Unit and will continue to operate.

The operations staff is trained and is sensitive to the need to get SW back as soon as possible, as it eases event recovery by making BOP systems available for use during the recovery. Restoration of the SW system could begin as early as 1 hour following a LOCA event and could take from 4-6 hours. The service water system restoration activities, except for venting of the fuel pool cooling heat exchangers and reactor building chillers, do not require reactor building access. Venting of these heat exchangers is done for optimizing system performance, not for waterhammer protection. Considering the potential post accident environment in the reactor building, optimization of fuel pool cooling heat exchanger and reactor building chiller operation

may occur at some later time. Area radiological assessments would be performed to determine accessibility. The recovery would be accomplished through implementation of procedures ON-135-001 "Loss of Fuel Pool Cooling/Coolant Inventory", ON-111-001, "Loss of Service Water", and OP-111-001, "Service Water System". The principal actions to be taken to restore cooling to the SFPC system are:

1) ON-111-001 "Loss of Service Water" would be entered upon loss of service water. It is not expected that ON-111-001 would be entered through ON-135-001 (Loss of Fuel Pool Cooling) since restoration of SW is desired for restoration activities that would occur prior to the need to reestablish fuel pool cooling. Instruction 3.3.2 of ON-111-001 "Loss of Service Water" requires checking of the Service Water pump supply breakers for trips and lockouts. The breakers are located on the 699' elevation of the turbine building. The lockout relays would then be reset. This activity is not expected to take place prior to 1 hour after the LOCA (to assure ECCS pump voltage would not be affected). The ON then directs pump restart to occur per OP-111-001 "Service Water".

2) With power now available to the pumps, Section 3.1 "Setup of Service Water System for Normal Operation" in OP-111-001 "Service Water System" would be implemented. Note that once it is confirmed that the pumps are operating satisfactorily, the various user chillers and heat exchangers are to be vented to assure optimal heat exchanger and chiller performance in Section 3.1.25. Venting of all heat exchanger and chillers would not be required since some are below the cooling tower basin elevation. The fuel pool cooling heat exchangers and reactor building chillers would be expected to need venting for optimum system performance. In order to perform this venting activity for the fuel pool cooling heat exchangers, the reactor building would be entered. These heat exchangers are located on the 749' elevation. This venting would take 5-10 minutes per heat exchanger. The reactor building chillers are also located in the reactor building on 749' and would be vented. It would take between 4 - 6 hours to implement Section 3.1 of the OP-111-001.

The only activities listed above that require access to the reactor building are the venting of the SFPC heat exchangers and the Reactor Building chillers.

The dose the operator would receive venting the fuel pool cooling heat exchangers has not specifically been determined as it is largely dependent upon when the venting action is taken. However, it is expected to be less than the maximum dose calculated to be received by an operator restoring SFPC as they are in the same general plant location. This dose has been calculated and is included in PLA-4069 dated 1/4/94.

Note that the preceding evaluation discusses the Unit 1 response and procedures. Unit 2 response would be identical except Unit 2 procedures would be utilized.

RESPONSE TO NRC QUESTION 1b

As noted in the response to question 1a, the SFP cooling system would continue to operate following a LOCA without a LOOP and a loss of service water. The operators are not procedurally directed to trip the SFPC pumps following a loss of service water, unless service water cannot be restored. Since this question is concerned with restoration of service water, the actions for loss of service water without restoration of SFPC system will not be considered.

Since the SFPC system remains in operation, the only actions required will be periodic make-up to compensate for evaporative losses. Make-up would not be necessary until after service water is restored. An evaluation of the impact of operation of the SFPC system without cooling capability is provided below.

TIME ESTIMATE - FUEL POOL MAKEUP

For this evaluation, the following is assumed :

1) ESW is the only available makeup source.
2) Fuel Pool Cooling system is operating service water is not, thus no pool cooling is occurring.
3) Isolated fuel pool
4) LOCA
5) The pool decay heat will be assumed to be 8.2 MBTU/HR. This is the heat load utilized in the evaluation supporting response 2 in PLA- 4069 dated 1/4/94.
6) The evaporation rate is 5787 lbm, which is conservatively based on a heat load of 12.6 MBTU/HR.

OP-135-001 R17 and OP-235-001 R16 (Fuel Pool Cooling) identify in section 3.2.7 that skimmer surge tank level should be maintained between 67% to 90%. Operators make-up approximately 10% skimmer surge tank level once per shift, to maintain level at 90%. It will therefore be assumed that the tank is filled to the 90% level and 12 hours later the event occurs before makeup is provided with 10 % evaporative loss occurring. Thus, the tank level is 80 % full.

Until the Service Water system (SW) is restored to service, the Fuel Pool Cooling system pumps would remain in operation. The fuel pools would heatup due to the decay heat from the fuel and from the operating fuel pool cooling pumps. The heat added to the fluid by the operating pumps is the friction horsepower (FHP). FHP is the difference between the brake horsepower (BHP), which is the power delivered to the pump shaft, and the hydraulic horsepower (WHP), which is the pump output. The pump efficiency (N_p) is equal to WHP/BHP. From pump curves for the 1P211A and 1P211 B pumps at 600 GPM flowrate, it can be seen that BHP = 50 and N_p = 60 %.

Based on this information, the heat added by the SFPC pumps is 152,700 BTU/HR. This is then added to the SFP decay heat of 8,200,000 BTU/HR, which greatly exceeds the heat added by the pumps. This combined heat load will result in a temperature increase of 67.2°F over the time period between the loss of Service Water and its restoration. Thus, the SFP will at most peak at 177°F.

During this heatup period of time, the pool evaporation will increase. Calculations shows that the fuel pool level and skimmer surge tank level would be effectively constant due to the thermal expansion effects of the heatup. Once service water was started and cooling restored, the level would drop due to thermal contraction. Assuming this contraction occurred instantly down to the pool temperature at the start of the event, the level loss would be 700 gallons. This will result in a reduction in tank level of 25.7 inches, or 9.7 % of tank level.

Using 10 % /12 hrs to be conservative; skimmer surge tank level drop as a function of time is:

Time(hours)	Indicated Tank Level	Activity
-12	90%	Makeup provided.
0	80%	Event occurs prior to makeup.
12	70%	Begin restoring Service Water
15	66%	Low level alarm
24	60%	Service Water Restored
82	11%	Pump trip setpoint.

The low level alarm is set for 66% (193"). The low level pump trip for NPSH pump protection occurs at the 11% level (30").

Based on the above, ample time exists to provide makeup prior to reaching the low level pump trip which would not occur earlier than 82 hours after event occurrence. Therefore, actions for providing make-up via ESW are not required until 24 to 82 hours.

QUESTION 2

In previous discussions with the staff, including discussions on July 8, 1993, PP&L indicated that one possible action for mitigation or preventing the spread of vapor from one (or two) boiling spent fuel pools entails shutting down the reactor building recirculation fans. Please describe your plans to provide procedures to perform these activities including guidance to the emergency response staff and/or system-specific procedures.

RESPONSE TO QUESTION 2

EP-PS-102 "Technical Support Coordinator: Emergency Plan-Position Specific Procedure" Tab I presently addresses the Fuel Pool Boil event. The major task of this section is to "Determine if Fuel Pool boiling can be expected and initiate actions as necessary to prevent Fuel Pool boiling or to mitigate the consequence of Fuel Pool boiling." This procedure will be revised to reflect the following:

1. If a loss of fuel pool cooling event occurs and cooling cannot be restored, yet no source term is present (ie, seismic event with a LOOP, both fuel pools expected to boil, makeup available) then the following actions will be required.

   (a) Isolate Zone(s) 1 and 2 from the Recirculation Plenum to preclude any spread of high temperature/high humidity environment to other areas of the Reactor Building, and
   (b) Shutdown the Reactor Building Recirculation and SGTS fan(s), and
   (c) Vent the refueling floor directly to atmosphere.

   Actions (a) and (b) will be accomplished in the Control Structure.
   Action (c) will be accomplished in the Reactor Building.

2. If a loss of fuel pool cooling event occurs and cooling cannot be restored and a source term is present (ie; LOCA/LOOP, one fuel pool expected to boil, makeup available) then the following actions are required.

   (a) Shutdown the Reactor Building Recirculation Fan(s), and
   (b) Maintain SGTS in service, and
   (c) Any zone that automatically aligned to the Recirculation Plenum shall remained aligned.

   Action (a) will be accomplished in the Control Structure.

The above procedure changes require additional evaluations to determine the appropriate time to turn off the recirculation fans. Due to COTTAP modeling constraints it will take approximately 4 months to complete the code revision and perform the analysis. It will take an additional month to prepare and approve the procedure change following the analysis.

Therefore, PP&L commits to completing the procedure changes identified above by October 1, 1994.

QUESTION 3

The staff has reviewed additional portions of the licensing basis documentation. Section 9.2.5 of the Final Safety Analysis Report (FSAR) describes the emergency service water system (ESW) and Section 3.1.2.4.15 describes compliance of the facility design with General Design Criteria 44. Section 3.1.2.4.15 states:

> "The emergency safeguard service water system, which comprises both the Emergency Service Water System and the Residual Heat Removal Service Water system, provides cooling water for the removal of excess heat from all structures, systems and components which are necessary to maintain safety during all abnormal and accident conditions. These include the standby diesel generators, the RHR pump oil coolers and seal water coolers, the core spray pump room unit coolers, [reactor core isolation cooling] RCIC pump room unit coolers, the [high pressure coolant injection] HPCI pump room unit coolers, the [residual heat removal] RHR heat exchangers, RHR pump room unit coolers, emergency switchgear and load center room coolers, the control structure chiller and the fuel pool make-up."

Section 9.2.1 of the SER describes the above function of the ESW system and states:

> "The emergency service water system is an engineered safety features system designed to supply cooling water to the emergency diesel generators, residual heat removal pumps and to those rooms identified below that are required during normal and emergency conditions to safely shutdown the plant. The emergency service water system takes water from the spray pond (ultimate heat sink), pumps it to the heat exchangers which serves the above components or systems and returns it to the spray pond by way of a network of sprays.
>
> The emergency service water system is required to supply cooling water to the residual heat removal pumps room coolers, residual heat removal pump bearing oil coolers,... and to the spent fuel pools as emergency makeup...
>
> ...Therefore a failure of the nonsafety related piping coupled with any single active failure of the safety-related Emergency Service Water System will not preclude one of the loops from performing its function. By providing this isolation capability and redundancy in components, we conclude that the requirements is General Design Criteria 44 "Cooling Water" are met, including the single active failure criterion."

The staff concludes that provision of make-up to the spent fuel pool during a design basis accident, including design basis loss-of-coolant accident, is within the design and licensing basis of the SSES facility. The staff review did not conclude that boiling of the spent fuel pool was necessarily implied but recognizes that make-up to the pool will be necessary to compensate for, at the very least, evaporative losses.

The staff requests that you provide information to demonstrate that make-up can be supplied to the spent fuel pool from the ESW system under design basis accident conditions (Specifically, design basis LOCA). The staff has noted the dose estimates that you supplied in previous correspondence regarding activities necessary to manipulate the ESW-fuel pool make-up valves under various accident conditions. The staff requests that you reanalyze the performance of these activates using design basis assumptions. Please specifically describe the correlation between the point in the accident sequence time line that make-up actions would be required, and times in the accident sequence when expected operator dose would exceed design basis limits.

RESPONSE TO QUESTION 3

A radiation dose analysis was performed to evaluate personnel access doses inside the reactor building for providing Emergency Service Water (ESW) make-up to the spent fuel pool under DBA-LOCA accident conditions, without a LOOP. This analysis evaluates the adequacy of the reactor building radiation shielding design and addresses the operator access doses from contained radiation sources. Consistent with SSES FSAR Chapter 18.1.20, which was performed as a required response to Item II.B.2 of NUREG-0737, post-LOCA airborne radiation doses are not addressed in this calculation.

Operator access doses are calculated and based on point specific dose rates which are determined from the actual locations of the radiation source terms and the proximity of operator access routes in relation to those sources. Detailed access dose analyses were performed at 24 hours post-LOCA for both ESW tie-in and ESW flow control missions. Access doses at other time periods were then evaluated by multiplying the dose results at 24 hours by the ratio of the radiation source term at the time period of interest to the source term at 24 hours.

Analyses were performed to determine the time post-LOCA at which ESW make-up should be initiated. It was determined that access can occur as early as 24 hours after the DBA-LOCA, and as late as 82 hours post-LOCA. To evaluate the accident sequence time line, 3 times were evaluated: 24 hours post LOCA, 40 hours post LOCA and 82 hours post LOCA.

The mission dose for access to Elevation 670' of the reactor building to tie-in the ESW system for make-up to the spent fuel pool was determined to be 7.27 Rem at 24 hours post-LOCA. For operator entry at 40 hours post-LOCA, using the scaling procedure described above, the mission dose was found to be 4.8 Rem. If entry is delayed until 82 hours post-LOCA, the mission dose decreases to 2.75 Rem.

The operator mission dose for access to Elevation 749' of the reactor building to control ESW make-up flow to the spent fuel pool at 24 hours post-LOCA is 1.41 Rem. Note that this is a separate and later action required after ESW tie-in has occurred. It will also be performed on a periodic basis for the duration of the event. Although not specifically evaluated at 40 and 82 hours, mission doses for control of ESW makeup to the spent fuel pool would decrease for time periods greater than 24 hours post-LOCA.

The dose acceptance criteria in NUREG-0737, Item II.B.2 states that doses should not exceed 5 Rem to the whole body or its equivalent. With this limit in mind, and if conditions warrant, operator access to provide ESW make-up to the spent fuel pool should be delayed at least until 40 hours. Ample skimmer surge tank capacity exists to allow delaying operator access for make-up in order to minimize dose. However, this restriction must be weighed by a prudent ALARA review, and if necessary, entry earlier than 40 hours should be expected.

Susquehanna is designed such that initiation of ESW make-up to either SFP results in both pools being filled regardless of whether or not the pools are cross-tied. With the pools isolated, water added to one pool will overflow to its skimmer surge tank; which when completely filled will overflow to the cask storage pit; which will overflow to the opposite units' skimmer surge tank; which will in turn overflow to the opposite units SFP. Therefore, operator access to the LOCA unit is not required to assure make-up to the LOCA unit SFP and doses evaluated in this response would not be experienced.

The following summarizes the assumptions and data used in the above mission dose evaluation:

1. This analysis is based on Design Basis Loss-Of-Coolant Accident (DBA-LOCA) conditions.

2. The activity source term for this analysis is based on the requirements on NUREG-0737, Item II.B. for post-LOCA liquid containing systems and is 50% of the core equilibrium halogen activity inventory and 1% of the core equilibrium particulate activity inventory released to the suppression pool water. Post-LOCA airborne radiation sources are not considered.

3. The equilibrium core inventory is based on power uprate conditions. The SSES power uprate core thermal power level for Design Basis Accident Analysis is 3616 MWt which is 105% of the uprated core thermal power.

4. Operator access doses are based on point specific dose rates which are determined from the actual locations of the radiation source terms and the proximity of operator access routes in relation to those sources. Detailed analyses were performed at 24 hours post-LOCA. Access doses at other time periods were evaluated by multiplying the dose results at 24 hours by the ratio of the radiation source term at the time period of interest to the source term at 24 hours.

5. In order to provide ESW makeup to the spent fuel pool, operator access is required to valves in the Fuel Pool Cooling System to initially tie-in flow from the ESW system and then to control the ESW makeup flow rate.

   System Tie-in:

   UNIT 1: Open either valve 153500 or 153501.
   UNIT 2: Open either valve 253500 or 253501.

   ESW Makeup Flow Control:

   UNIT 1: Control flow with valves 153090A&B or 153091A&B
   UNIT 2: Control flow with valves 253090A&B or 253091A&B

   Note: For flow control, both the 090 and 091 valves must be opened in each ESW supply line being used for makeup.

6. Two separate operator access missions are assumed in order to provide ESW makeup to the spent fuel pool under LOCA conditions. One operator access mission is required to tie-in the ESW system to the spent fuel pool and is a one time access requirement. The second operator access mission is required to control ESW system makeup flow and is a periodic access requirement following system tie-in. These missions are as follows:

Access for System Tie-In

One-time operator access to the following areas is required to tie-in the ESW system for makeup to the spent fuel pool:

UNIT 1: Reactor Building Elev. 670'-0", Equipment Area, Room # I-105

Open valve 153500 and 153501.

UNIT 2: Reactor Building Elev. 683'-0", Closed Cooling Water Heat Exchanger/Pump Room, Room # II-203

Open valve 253500 and 253501.

ESW makeup to the spent fuel pool can be provided by opening either valve 153500(253500) or 153501(253501). It is conservatively assumed for this dose analysis that both valves are opened by the operator during access to elevation 670' of the reactor building in order to make both ESW loops available for supplying makeup to the spent fuel pool.

Access for Flow Control

Following ESW system tie-in, periodic operator access to the following areas is required to control ESW system makeup flow to the spent fuel pool:

UNIT 1: Reactor Building Elev. 749'-1", Fuel Pool Cooling Heat Exchanger Pump Room, Room # I-514

Only one ESW loop is required to provide make-up to the spent fuel pool. Open fuel pool cooling and clean-up valves 153090 and 153091 in one ESW supply loop. Control flow with valves 153090 or 153091 in this loop.

UNIT 2: Reactor Building Elev. 749'-1", Fuel Pool Cooling Heat Exchanger Pump Room, Room # II-514

Only one ESW loop is required to provide make-up to the spent fuel pool. Open fuel pool cooling and clean-up valves 253090 and 253091 in one ESW supply loop. Control flow with valves 253090 or 253091 in this loop.

7. Plant operating procedures require that makeup to the spent fuel pool be made for skimmer surge tank levels between 67% and 90%. Following a DBA-LOCA, the sequence of events for losses from the spent fuel pool are:

Time(hours)	Indicated Tank Level	Activity
-12	90%	Makeup provided.
0	80%	Event occurs prior to makeup.
12	70%	
15	66%	Skimmer surge tank low level alarm setpoint.
24	60%	
82	11%	Pump trip setpoint.

The skimmer surge tank low level alarm is set at the 66% water level. The low level pump trip for NPSH pump protection occurs at the 11% water level. Ample time exists to provide makeup prior to reaching the low level pump trip which would not occur until approximately 82 hours after event occurrence. Therefore operator access to provide ESW make-up is not required prior to 24 hours post-LOCA, but must be completed within 82 hours post-LOCA. Thus, Operator access doses are evaluated at 24 and 82 hours post-LOCA.

8. A time motion study was performed to determine operator access and travel times inside the reactor building under LOCA conditions. An operator was dressed in protective clothing and wore a Self Contained Breathing Apparatus and actual transit times to valves located on elevations 670' and 749' of the reactor building were measured. Results of this time motion study are used to evaluate operator access doses.

9. Dose acceptance criteria for personnel access to a vital area during the course of an accident is given in NUREG-0737, Item II.B.2 and states that the doses should not exceed 5 Rem whole body or its equivalent to any part of the body for the duration of the accident.

QUESTION 4

Earlier PP&L submittals have identified that a SFP with no operable cooling system may be cooled by natural convection through the cask storage pit by cooling systems associated with the remaining SFP. The staff requests that PP&L describe the basis (i.e., test results, calculational results, or operational experience) for concluding that this method of cooling is adequate. The staff also requests that PP&L estimate the temperature difference between the pools assuming that the decay heat load in the SFP without an operable cooling system is 6.2×10^6 BTU/hr.

RESPONSE TO QUESTION 4

In earlier submittals to the NRC, PP&L has identified that a SFP with no operable cooling system is adequately cooled by natural circulation through the cask storage pit using cooling from the other SFP. However, upon further review PP&L has determined that the outage unit's service water system is shutdown, but the SFPC system remains in operation during outage periods. Therefore, even though no cooling is provided by the outage unit's SFPC system, it will aid in mixing the outage unit's SFP. The ability of the non-outage SFPC system to cool both SFPs is based upon plant operational data taken during refueling outages. The outage operations are controlled by TP-135(235)-011 which maintains the SFPC pumps in operation. The SFPC operating procedure, OP-135(235)-001 directs the operator to shutdown and isolate the SFPC system of a unit when the other unit is providing cooling to both SFPs via the cask storage pit. While the configuration identified in the operating procedure is not benchmarked against plant operational data, PP&L believes that the results would be similar to those obtained during outages. The basis for this is explained in more detail below.

Test procedure TP-135(235)-011 is performed at each refueling outage to monitor fuel pool temperature and heat load. As noted above, this procedure only isolates Service Water to the SFPC heat exchangers and keeps the outage unit's SFPC pumps in operation. The data has been recorded for the past three (3) refueling outages and indicates that the temperature difference between the two SFPs has generally been less than one degree (1°F) throughout the duration of the outage. The in-plant tests have shown that this temperature difference between the two pools can be maintained with a heat load of approximately 20×10^6 Btu/hr in the spent fuel pool without an operable cooling system. This data demonstrates that adequate thermal mixing occurs between the pools via natural circulation.

No significant difference is expected for lower heat loads. Mixing is inherent in the geometry of the fuel pool as the heated water rises from between the spent fuel bundles and mixes with the fluid above. The water exiting the spent fuel region is replaced by cooler fluid drawn down between the fuel and the pool walls, and a natural circulation flow is established. While a lower heat load will reduce the magnitude of the velocities induced by natural circulation through the spent fuel bundles, mixing will continue due to the nature of buoyancy and the hotter fluid will continually rise towards the free surface. The fuel pool cooling system takes a suction on the spent fuel pool via a weir at the free surface, drawing the warmest water from the pools. The in-plant tests have shown that this arrangement can effectively draw the hotter fluid from the opposite pool. A lower heat load will not affect the potential of the skimmer arrangement to draw fluid towards itself. As fluid is skimmed off of the surface, buoyancy acts to maintain a hot layer of essentially uniform thickness among all pools in communication. The temperature of one pool will not get significantly hotter than the other as long as the fluid near the top of both pools is in communication. Buoyancy will act to drive the hotter fluid to a stratified layer

from which the skimmer will draw its suction. The same physical process will result at lower heat loads. Therefore, the temperature difference between the pools with a decay heat load of 6.2×10^6 BTU/hr in the spent fuel pool without an operable cooling system is expected to be on the order of 1 degree Fahrenheit.

During the in-plant tests, both fuel pool cooling systems were in service, however, the cooling function of the reload unit's pool was removed by shutting down service water flow to its heat exchangers. The reload unit's fuel pool cooling system provided some mixing effect but no cooling to its pool. Operation of the fuel pool cooling pumps on the reload unit recirculated a portion of the fluid within the pool keeping it from migrating to the opposite pool for cooling. While one of the fuel pool cooling return lines discharges towards the cross-tie point and could contribute to fluid transport to the opposite pool, the fluid must travel the width of the pool to reach the cross-tie point and the lateral velocity would diffuse allowing the fluid to be drawn downward by the natural circulation patterns. In addition, the fuel pools are a mirror image of each other with respect to the cross-tie point and the discharge momentum imparted to the fluid in both pools would tend to cancel each other. The less internal recirculation there is in the reload unit's fuel pool the more influence the opposite unit's cooling system can have on it and the greater the potential for fluid transport between the pools. Therefore, the cooling capability demonstrated by the in-plant tests should also result with no fuel pool cooling flow on the reload unit.

QUESTION 5

Bechtel specifications for the watertight doors between the "A" core spray/reactor building sump room and other ECCS pump rooms reviewed during an audit on February 7, 1994, indicated that an unseating pressure of zero was specified for certain water tight doors. However, PP&L Calculation EC-035-0510 stated that the relevant watertight doors provide protection to 15 psid based on a pre-delivery hydrostatic test. Because the subject watertight doors were credited with preventing flooding of ECCS pump rooms in certain analyses, the staff requests that PP&L clarify the apparent inconsistency between the specification and the assumed performance of the watertight doors in Calculation EC-035-0510.

RESPONSE TO QUESTION 5

During fuel pool boiling scenarios, condensate from the 818' elevation would drain to the basement of the reactor building. The areas into which this flow would drain are defined by the outline of the respective Unit's reactor building sump and "A" Core Spray pump rooms. Note that although there is a fire door separating these two areas, it is not watertight, and hence both areas would flood concurrently.

Watertight doors are located between the ECCS pumps rooms on the 645' elevation to provide for flooding protection. These doors were procured for the Susquehanna Units under Bechtel Specification No. 8856-A-16 (Watertight Doors). In this document, both seating and unseating pressures for each door were specified. In addition, the specification also required that each door be designed and fabricated to withstand a test pressure of 1.25 times the specified pressure. Table 1 identifies the doors affected during the fuel pool boiling scenario, along with the appropriate information from Specification 8856-A-16.
From this table, it is seen that the lowest specified seating pressure is 10 psid, which corresponds to a water height of 23 feet. Therefore, this was the maximum allowable height of water used in the reactor building flooding assessment (PP&L Calculation EC-035-510, Revision 1).

In order for Door #s 25 and 26 to contain the area flooded as described above, they are required to prevent leakage in the unseated direction. From the data in Table 1, it is seen that the original specification did not specify this as a requirement. However, all doors procured and delivered under this purchase order are of identical design.

The only difference between the doors which were required, or not required to be leak-tight in the unseated direction is the scope of post fabrication testing. All doors which were specified to be leak tight in both directions were hydrostatically tested in both directions, while those that were only specified to be leak tight in the seating direction were tested as such.

The original hydrostatic test reports were reviewed and indicate that the design of these doors is capable of withstanding 15 psi (1.25 x 12 psi) in both directions. Although the original purchase specification does not require that Door #s 25 & 26 be leak-tight in the unseating direction, they are, by design, none-the-less watertight in both directions for pressures up to 15 psi (34.5 feet of water).

As a result, it is both conservative and reasonable to use a maximum flood height of 23 feet in both directions.

Table 1 - Watertight Doors Rated Pressure (psi)

Unit	Door #	Areas Separated	Seating Press	Unseating Press
1	13	CS "A" Rm / CS "B" Rm	10	10
1	25	RB Sump Rm / RHR "A" Rm	12	0
2	12	CS "A" Rm / Stair 202	10	0
2	14	CS "A" Rm / CS "B" Rm	10	10
2	26	RB Sump Rm / RHR "A" Rm	12	0

QUESTION 6

The staff requests that PP&L evaluate the qualification of standby gas treatment system (SGTS) components within the control structure for operation in the environment created by ventilating the reactor building through the system for cases where one SFP is boiling and where two SFPs are boiling. The assumptions used in the evaluation should be consistent with previous evaluation of SGTS duct conditions during SFP boiling scenarios.

RESPONSE TO QUESTION 6

The qualification of standby gas treatment system (STGS) components have been analyzed for the case with one SFP boiling (LOCA/LOOP) and for the case with two SFPs boiling (seismic event).

LOCA/LOOP

PP&L has analyzed the impact on equipment qualification of higher room temperature due to loss of SFPC during a LOCA/LOOP. This case is based on Compartment Transient Temperature Analysis Program (COTTAP) temperature inputs. The conclusion was that temperatures in the Control Structure SGTS rooms do not exceed 104°F. All equipment in these areas is qualified for 104°F or higher in the Environmental Qualification Program.

Seismic Event

PP&L has performed evaluations similar to the LOCA event for a seismic event. The conclusion was that, using a very conservative analysis, temperatures in the Control Structure SGTS room will not exceed 104°F at the point in time when SGTS capability will be lost due to condensation in the recirculation plenum as discussed in PLA-4133, dated 5/4/94. All equipment in these areas is qualified for 104°F or higher in the Environmental Qualification Program.

QUESTION 7

In the March 7, 1994 RAI, the staff noted that the licensing basis of the plant credits the use of the SGTS during the boiling pool event following a seismic event. In the RAI, the staff asked a question to assess the potential to use the residual heat removal system to cool the spent fuel pools following a seismic event. The diesel generator loading associated with a seismic event and coincident loss of offsite power (LOOP) was presented in PP&L's letter dated March 25, 1994. The staff requests that PP&L perform an assessment of diesel generator loading considering the limiting single failure for a seismic event with a coincident LOOP. Justification for the assumed single failure should be provided.

The staff also requests that PP&L assess the ability of the diesel generators to accommodate the additional loading associated with operating one loop of the RHR system in the SFP cooling assist mode and one loop of alternate decay heat removal, as described in procedure ON-1(2)49-001, "Loss of RHR Shutdown Cooling," on the non-accident unit. The intent is to determine maximum diesel generator loading during a loss of coolant accident (LOCA) in the opposite unit coincident with a LOOP, assuming no single failure. If one of the diesel generator loading patterns presented in the FSAR is bounding, describe the basis for this conclusion.

RESPONSE TO QUESTION 7

This evaluation discusses the plant response to a seismic event concurrent with a LOOP, and the subsequent loss of Spent Fuel Pool Cooling (SFPC). Details regarding how the plant will be placed in safe shutdown for a seismic event are also provided. Such an evaluation is needed to determine the single failures that would impact use of RHR in the Fuel Pool Cooling mode concurrent with maintaining safe shutdown of both reactors. During such an event, the SFPC is assumed to fail due to the earthquake loading resulting in pipe stresses over allowables for both the SFPC and service water systems. It will also be unavailable due to the LOOP. Consequently it will be necessary to restore cooling prior to boiling of the SFPs in order to avoid the impact on SGTS discussed in PLA-4133, dated 5/4/94. Therefore, this evaluation only discusses the plant response and operator actions required to restore cooling to the SFPs via the RHR Fuel Pool Cooling Mode, and place both units in cold shutdown without exceeding any design limits.

The following assumptions are made for this analysis:

1. A Safe Shutdown Earthquake causes a LOOP which results in an MSIV closure event.

2. No Seismic Category I equipment fails as a result of the earthquake.

3. Seismic category I equipment is subject to random failures independent of the earthquake. Single failures are assessed per ANSI/ANS-58.9-1981.

4. Equipment other than seismic category 1 equipment fails as a result of the earthquake.

5. Operator actions including manual in plant equipment manipulations are allowed provided that they are performed no sooner than 10 minutes after the earthquake and sufficient time is available to reliably execute the procedure.

6. Plant Configurations:

   1. Both units in power operation without pool communication

   2. Both units in power operation with pool communication,

   3. One unit at power and the other unit in refueling without pool communication.

   4. One unit at power and the other unit in refueling with pool communication.

7. Operating Restrictions:

   1. Shutdown cooling and fuel pool cooling cannot be operated simultaneously on the same unit.

   2. Division I of RHR cannot operate in suppression pool cooling with division II of RHR in fuel pool cooling.

   3. Division I of RHR can operate in fuel pool cooling with division II RHR in suppression pool cooling.

The initiating event is an earthquake at time zero. The earthquake is assumed to cause a Loss of Offsite Power (LOOP). The immediate response of the plant to a LOOP includes:

- reactor scram following loss of power,
- MSIV isolation,
- safety relief valve actuation following the MSIV isolation,
- condensate/feedwater pump trip on loss of power and low suction pressure,
- low low RPV water level following void collapse
- reactor building isolation and SGTS initiation.

The above events lead to the following:

- auto initiation and loading of the diesel generators,
- auto initiation of RCIC and HPCI,
- operator entry into the following procedures:

SCRAM, ON-100/200-101
RPV Control, EO-100/200-102
Primary Containment Control, EO-100/200-103

The operator enters the above procedures based upon any of the following conditions: a scram condition, the RPV water level less than +13", the RPV pressure greater than 1037 PSIG, the drywell pressure greater than 1.72 PSIG and the suppression pool temperature greater than 90°F. Entry into the Secondary Containment Control Procedure EO-100/200-104 is not expected until after 12 hours when the average reactor building temperature reaches 110°F. If offsite power is not lost or recovered prior to the reactor building temperature exceeding the maximum normal temperature, the secondary containment procedure will not be entered. The operator will execute these procedures concurrently. The more significant actions taken are described below:

SCRAM ON-100/200-101, Rev 1

- ensuring a scram and the appropriate containment isolation,
- ensuring diesel generator initiation and loading,
- ensuring ESW initiation and proper operation,
- placing the mode switch into shutdown.

The above actions are performed immediately.

RPV Control EO-100/200-102, Rev 5

Level Control

- Restoring and maintaining the RPV water level between +13" and +54" using RCIC, HPCI, Core Spray and RHR as needed,
- Resetting the generator lockout.

Pressure Control

- Preventing automatic actuation of SRVs by manual operating the SRVs and operating HPCI in the pressure control mode,
- Maintaining the RPV pressure below the HCTL curve,
- Depressurizing the RPV at less than 100°F/hr.

Once the level is controlled between the band +13 to +54 and the shutdown cooling interlocks have cleared, the procedure allows the operator to establish shutdown cooling.

Primary Containment Control EO-100/200-103, Rev 5

- Restoring and maintaining the suppression pool below 90 °F using suppression pool cooling,
- Maintaining the RPV pressure and the suppression pool water level below the SRV tailpipe level limit.

The operator will take immediate action to restore the reactor and containment parameters to normal conditions. A significant amount of time exists to successfully perform these actions. Once the operator has ensured a successful scram has taken place, the first actions are to ensure the RPV water level and pressure are being controlled. The RCIC and HPCI systems will automatically initiate and quickly flood RPV. Prior to the RPV water level reaching +54", the operator will place the HPCI system in the pressure control mode to avoid RCIC and HPCI trip on high water level, and to augment the SRVs for RPV pressure control. After about 2 hours the HPCI system is sufficient to control RPV pressure without the use of the SRVs. The HPCI system can be used for up to 3.5 days to control RPV pressure if necessary. If the water level is in the normal range the operator will commence a slow RPV depressurization at less than 100 °F/hr. Once the shutdown cooling interlocks have cleared the operator is permitted to place the unit into shutdown cooling or alternate shutdown cooling[1]. These interlocks should not be cleared prior to 3 hours.

In the unlikely event that the RCIC and HPCI systems are both unavailable, the operator has 40 minutes to restore them to operation prior to having to initiate an emergency depressurization to allow either the core spray or RHR systems to be used for core cooling. Once RPV water level is maintained above +13", the operator may enter shutdown cooling.

The operators will also be controlling the primary containment parameters, especially the suppression pool temperature and water level. The operator will initiate a loop of suppression pool cooling within the first 30 minutes. However the operator has over 8 hour to initiate suppression pool cooling before exceeding the pool HCTL, which is the first critical parameter to be encountered.

Once RPV and Primary Containment control is established, the operator will then establish fuel pool cooling. For this scenario, if fuel pool cooling is established within 35 hours[2], fuel pool boiling will be avoided. The operator will first attempt to establish normal fuel pool cooling. If this system is unavailable due to the unavailability of offsite power or seismic induced damage, the operator will then align the RHR system in the fuel pool cooling assist mode. It is estimated that this alignment will require about 8 to 12 hours. A single loop of RHR in the fuel pool cooling assist mode is more than capable of removing heat from both fuel pools, provided that the pools are connected.

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[1] Per GO-100/200-005 or ON-149/249-001

[2] Attachment to PLA-4133, dated 5/4/94

PLANT CONFIGURATION EVALUATION

Evaluation of Configuration 1:
Both units at power; operation without fuel pool communication.

Review of the fuel pool risk assessment calculation[3] shows that there are 7 single failures in the RHR fuel pool cooling assist mode of operation. Additionally, loss of a single diesel will also prevent simultaneous operation of SDC and fuel pool cooling on both units. Therefore this case plus a single failure will lead to one of the two fuel pools boiling or a unit being unable to be placed into shutdown cooling.

Case Plant configuration 3 has a lower heat load, however the systems analysis and results are equivalent to configuration 1. Therefore, configuration 3 is not evaluated independently.

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[3] Probability of Fuel Pool Boiling, SA-TSY-001,11/23/93

Evaluation of Configuration 2:
Both units at power; operation with fuel pool communication.

A review of the plant systems required to simultaneously operate Shutdown Cooling (SDC) or Alternate SDC and cool the connected fuel pools was performed. This evaluation revealed that the diesel generators and ESW/Spray Pond Network were controlling with regard to single failure. It was concluded that a failure of a diesel generator can be tolerated, while the failure of a complete loop of ESW cannot. Additionally, the loading on the diesel generators with a single failure has been evaluated in a separate calculation and found to be within the continuous rating of the diesels. A summary of this calculation is provided as an attachment to this evaluation.

Both reactors and the fuel pools cannot be simultaneously cooled following an earthquake, if the spray pond bypass discharge valves, HV-01222A or HV-01222B fail to close when required by operator action. Closure of these valves is required to allow proper function of the spray pond spray network. It was determined based upon a review of ANSI/ANS-58.9-1981[4] that failure of these valves do not represent single failures.

These valves do not have to be closed until an RHR heat exchanger in the particular loop is placed into service. However, as stated above, operation of both these valves is not required until about 35 hours after the earthquake. Additionally, it is reasonable to expect that these valves can be manually closed or repaired prior to fuel pool boiling. They are located in the ESW valve vault which is located outside the reactor building in the site yard next to the spray pond. Furthermore the mean time to repair a valve is 5.2 hours[5]. Based upon these conditions these valves are excluded from single failure analysis per reference 5.

Therefore, both reactors and both crosstied fuel pools can be simultaneously cooled following a safe shutdown earthquake and a credible independent single failure (failure of a diesel generator).

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[4] American National Standards Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems, ANSI/ANS-58.9-1981. Feb. 17, 1981.

[5] NUREG/CR-3154

ATTACHMENT TO RESPONSE 7

DIESEL GENERATOR LOADING EVALUATION

WITH SINGLE FAILURE

SPENT FUEL POOL COOLING DIESEL GENERATOR LOADING

The purpose of this evaluation is to determine the Loading on the Diesel Generators for a Seismic Event with Loss of Spent Fuel Pool Cooling, an Extended Loss Of Offsite Power and single failure of one Diesel Generator at a time.

The following conditions were assumed for the attached Diesel Generator loading tables:

- Unit 1 and Unit 2 at 100% Power
- Seismic Event
- Loss of Unit 1 and Unit 2 Spent Fuel Pool Cooling
- Extended Loss Of Offsite Power
- Reactor cooling provided via ALTERNATE DECAY HEAT REMOVAL MODE
- Single Failure of one Diesel Generator at a time

The Diesel Generator Loading was developed using FSAR Table 8.3-1 for the equipment KW rating and the assignment of ESF and selected non-ESF loads to the Diesel Generators. The 4160 VAC cable losses were considered in the Diesel Generator Loading.

Separate Diesel Generator loading tables were developed for Control Structure HVAC Train "A" running with Train "B" in Standby; and Control Structure HVAC Train "B" running with Train "A" in Standby. This was done since only one train of Control Structure HVAC is running. Note FSAR Table 8.3-1a shows both trains of Control Structure HVAC running. This Table represents all loads connected to the Diesel Generator without regard to which loads are actually running.

Since the Diesel Generator Loading with the Control Structure HVAC Train "A" represents the most severe loading on the Diesel Generators, this loading was used as the base case for failure of the Diesel Generator A and B.

The loading tables were developed using Alternate Shutdown Cooling consisting of one RHR loop with one RHR pump in Suppression Pool Cooling and one Core Spray pump in Alternate Shutdown Cooling (See Procedure 0N-149-001). This method of achieving Alternate Shutdown Cooling was used instead of one RHR pump, because it represents the worst loading on the Diesel Generators.

For a single failure of a Diesel Generator the attached loading tables show that the Diesel Generators are capable of supplying the required loads to safely shutdown and maintain shutdown of Unit 1 and Unit 2 while the Fuel Pools are cooled by the Fuel Pool Mode of RHR. Also the continuous KW rating of the Diesel Generators are not exceeded for a single failure of a Diesel Generator.

The bounding condition for Diesel Generator Loading is as follows:

- Loading 0 - 10 minutes LOCA (FSAR Table 8.3-2,3,4,5)
- Loading 10 - 60 minutes Seismic Event (Attached tables 1,2,3,4)
- Loading beyond 60 minutes Seismic Event (Attached tables 1,2,3,4)

A summary of the total Diesel Generator Loading for both the LOCA and Seismic Events is found in attached table 5 .

For failure of the Diesel Generator A or B, the following ESW valves must be manually opened to support operation of Fuel Pool Cooling mode of RHR:

DG A Failure - HV-01222A, HV-01224A1, HV-01224A2

DG B Failure - HV-01222B, HV-01224B1, HV-01224B2

These valves are located outside the reactor building and are Seismic Category I.

ENCLOSURE 5

Joe Shea
After our phone call with Jack Hayes and Dave Powers from Sandia we found this letter. It appears to have all the info Dave requested. Suggest you forward to Jack.
Jim Kenny

PP&L

September 20, 1989

Ian B. Wall, Manager
Severe Accident Program
Nuclear Power Division
Electric Power Research Institute
3412 Hillview Avenue
P.O. Box 10412
Palo Alto, CA 94303

Dear Mr. Wall:

In response to your request regarding the constituents of concrete in various drywell floors, we have provided you with the attached information regarding the type of concrete that we have at our Susquehanna Steam Electric Station (SSES). In Appendix A of this document we have calculated the best estimates for mass fractions of various materials in concrete so that we can use them for modeling our containment using the code CONTAIN; we included Appendix A in this report since we thought you may find them useful.

We have not provided you with any of the references that we have used, however, we can provide you with a copy if needed. Should you have any questions please contact us.

Sincerely,

Shahin Seyedhosseini

SS/el
ss/msh1868c(28)

cc: C. A. Kukielka A2-3 ←
M. B. Detamore A2-3
P. R. Hill A2-3

<u>SSES Drywell Floor Concrete</u>: (from Reference 1 and Section 3.88 of the FSAR)

Class = C1

Design Strength = 4000 psi

Slump working limit at power of placement = 3" to 5"

Cement*: Type II Portland 15% Pozzolan by weight shall be used to replace cement in the mixes

Pozzolan* = class F (fly ash)

Coarse Aggregate*

Fine Aggregate* - Alluvial river sand

Admixtures* - Ligin or polymer type water reducing agent

Mix design: 12 cylinders casted for design
3 tested at age 90 days
3 tested at age 7 days
3 tested at age 28 days
3 tested at age 3 days

*Reference 2 shows the suppliers of these materials.

Drywell Concrete Constituents: (obtained from Reference 3 and corresponds to requirements of Reference 4 - see Appendix A for calculation)

Constituents	Weight (lbm) per cubic yard	% of total weight
Water (free from injurious amounts of oil, alkali, acid, organic matter)	286	7.52
Cement (Type II Portland)	460	12.1
Pozzolan (Class F. Fly Ash)	81	2.13
Coarse Aggregate	1865	49.05
Fine Aggregate (Alluvial river sand)	1109	29.166
Water Reducing Admixture	1.0125	0.026
Air Entraining Admixture	0.338	0.0090

$$\frac{\text{weight of water}}{\text{weight of water + cement}} = .53$$

Constituents of Cement (obtained from Ref. 6 and verifiable through Ref. 7)
Type II Portland

Constituents	(Actual) Per Ref. 6	(Required) Percentage
Silicon dioxide (SiO_2)	21.7	21.0 (min)
Aluminum oxide (Al_2O_3)	4.8	6.0 (max)
Ferric oxide (Fe_2O_3)	4.4	6.0 (max)
Magnesium oxide (MgO)	3.3	5.0 (max)
Alkalis (Na_2O+.658K_2O)	0.4	0.6 (max)
Sum of Tricalcium Silicate and Tricalcium Aluminate	38.5	58 (max)
Loss on Ignition (CO_2)	0.91	3 (max)
Insoluble Residue	0.41	0.75 (max)
Sulfur Trioxide (SO_3)	2.9	3 (max)
3 CaO-Al_2O_3	5.2	
Other	17.48	-

Constituents of Pozzolan (obtained from Ref. 5)
(Class F - Fly ash)

Constituents	% required
Silicon dioxide (SiO_2) + Aluminum oxide (Al_2O_3) + Iron oxide (Fe_2O_3)	70 (min)
Sulfur Trioxide (SO_3)	5 (max)
Moisture content	3 (max)
Loss on ignition (CO_2)	6 (max) (per Ref. 1)
Available alkalis as Na_2O	1.5 (max)
Magnesium oxide (MgO)	0% to 30.0% (max)

	% actual
SiO_2	45%
Al_2O_3 + Fe_2O_3	35%
MgO	1.5%
Mg, Na, K	6%
Loss on ignition (unburned carbon acids, SO_3, nitric)	3%
Others	9.5%

(Actual information is obtained from Ms. Kathy Shimp of PP&L)

Constituents of coarse aggregate (Per salesperson from Lycoming Silica Sand Co. (Mansdale Quarry))

Constituents	Percentage
$CaCO_3$	85.32
$MgCO_3$	7.53
$Fe2O_3$	0.34
Al_2O_3	0.212
SiO_2	0.086
Other	6.512

Constituents of Fine Aggregate
(obtained from Ref. 12)
(Alluvial river sand)

Constituents	Percentage
Chert (SiO_2)	2.3%
Sandstone[I] ($CaO + SiO_2$)	13.6%
Siltstone[II] (CaO)	33.5%
Quartz[III] (SiO_2)	50.6%

Note: The following definitions were obtained from a dictionary or an encyclopedia.

I Various colored sedimentary rock composed predominantly of sandlike quartz grains cemented by lime, silica, or other materials.
II Stone composed of hardened silt.
III A hard, crystalline, vitreous mineral silicon dioxide, SiO_2, found as a component of sandstone.

APPENDIX A

Best Estimates for Mass Fractions of Constituents of Concrete for use in CONTAIN

DETERMINATION OF CONCRETE TYPE FOR CONTAIN

For concrete the following % of composition can be derived (see next two pages):

$$SiO_2 = \underset{\text{cement}}{\underline{21.7\% \times 12.1\%}} + \underset{\text{coarse aggregate}}{\underline{.086\% \times 49.05\%}} + \underset{\text{pozzolan}}{\underline{45\% \times 2.13\%}}$$

$$+ \underset{\text{fire aggregate}}{\underline{(50.6 + 2.3)\% \times 29.166\% + 13.6/2\% \times 2.13\%}}$$

$$= .1920 + \underset{\text{cement}}{\underline{13\% \times 12.1\%}} = .2077$$

$$MgO = \underset{\text{pozzolan}}{\underline{1.5\% \times 2.13\%}} + \underset{\text{concrete}}{\underline{3.3\% \times 12.1\%}} + \underset{\text{coarse aggregate}}{\underline{3.58\% \times 49.05\%}} = .0219$$

$$CaO = \underset{\text{cement}}{\underline{5.2/2\ \% \times 12.1\%}} + \underset{\text{fine aggregate}}{\underline{13.6/2\ \% \times 29.166\% + 33.5\% \times 29.166\%}}$$

$$+ 47.78\% \times 49.05\% = .355 + 13\% \times 12.1\% = .3707$$

$$K_2O = .00024$$

$$Na_2O = \underset{\text{pozzolan}}{\underline{6\% \times 2.13\%}} = 12.1\% \times .2\% = .00154$$

$$Fe_2O_3 = \underset{\text{cement}}{\underline{4.4\% \times 12.1\%}} + \underset{\text{coarse aggregate}}{\underline{.34\% \times 49.05\%}} + \underset{\text{pozzolan}}{\underline{35/2\% \times 2.13\%}} = .0107$$

$$Al_2O_3 = \underset{\text{cement}}{\underline{4.8\% \times 12.1\%}} + \underset{\text{cement}}{\underline{5.2/2\% \times 12.1\%}} + \underset{\text{coarse aggregate}}{\underline{.212\% \times 49.05\%}}$$

$$+ \underset{\text{pozzolan}}{\underline{35/2\% \times 2.13\%}} = .0137 + \underset{\text{cement}}{\underline{13\% \times 12.1\%}} = .0294$$

$$CO_2 = \underset{\text{pozzalan}}{\underline{3\% \times 2.13\%}} + \underset{\text{cement}}{\underline{.91\% \times 12.1\%}} + (37.54\% + 3.94\%) \times 49.05\% = .2052$$

$$H_2O = .0752\%$$

According to Ref. 14 we have

Constituents of cement:

SiO_2 = 21.7%
Al_2O_3 = 4.8%
Fe_2O_3 = 4.4%
Na_2O+.658 K_2O = .4% (To come up with Table 1 we assumed .2% is Na_2O and 2% is K_2O)
MgO = 3.3%
CO_2 = .91%
SO_3 = 2.9%
CaO-Al_2O_3 = 5.2% (To come up with Table 1 we assumed 2.6% CaO and 2.6% Al_2O_3)
Ca_3Si_2 + Ca_3Al_2 = 38.5% (To come up with Table 1 we assumed 13% CaO, 13% Al_2O_3 and 13% SiO_2)
Other = 17.89%

Constituents of coarse aggregate:

$CaCo_3$ = 85.32% $\Longrightarrow$ CaO + CO_2 $\longrightarrow$ {CaO = 47.78%, CO_2 = 37.54%}
56 44

$MgCo_3$ = 7.53% $\Longrightarrow$ MgO + CO_2 $\longrightarrow$ {MgO = 3.58%, CO_2 = 3.98%}
40 44

Fe_2O_3 = .34%
Al_2O_3 = .212%
SiO_2 = .086%
Other = 6.512%

$CO_3Ca \longrightarrow CaO + CO_2$ {CaO = 56%, CO_2 = 44%}
100 56 44

$MgCO_3 \longrightarrow MgO + CO_2$ {MgO = 40/84 = 47.62%, CO_2 = 44/84 = 52.38%}
24+12+48 = 84; 24+16 = 40; 12+32 = 44

Consituents of Pozzolan:

$Al_2O_3 + Fe_2O_3 = 35\%$ assume 17.5% Al_2O_3, 17.5% Fe_2O_3
$SiO_2 = 46\%$
$CO_2 = 3\%$
$Mg + K + Na_2O = 6\%$ assume 2% Mg, 2%K, 2% Na_2O
$MgO = 1.5\%$
Other = 9.5%

Constituents of fine aggregate:

Sandstone (SiO_2 + CaO) = 13.6% assumes 6.8% SiO_2, 6.8% CaO

Siltstone (CaO) = 33.5%

Quartz (SiO_2) = 50.6%

Chert (SiO_2) = 2.3%

Constituents of concrete:

Water = 7.52%

Cement = 12.1%

Pozzolan = 2.13%

Coarse Aggregate = 49.05%

Fine Aggregate = 29.166%

Table 1

(Best estimates for Mass fractions of various componets in the SSES concrete for use in CONTAIN)

CONTAIN
VARIABLES

$fSiO_2$	(Mass fraction of SiO_2 in solid concrete)	= .2077 + .153=.3607
$fTiO_2$	(Mass fraction of TiO_2 in solid concrete)	= 0.0
fmno	(Mass fraction of MnO in solid concrete)	= 0.0
fmgo	(Mass fraction of MgO in solid concrete)	= 0.0219
fcao	(Mass fraction of CaO in solid concrete)	= 0.3707
fna2O	(Mass fraction of Na_2O in solid concrete)	= 0.00154
fk2O	(Mass fraction of K2O in solid concrete)	= 0.00024
ffe_2O_3	(Mass fraction of Fe_2O_3 in solid concrete)	= 0.0107
fal_2O_3	(Mass fraction of Al_2O_3 in solid concrete)	= 0.0294
fcr_2O_3	(Mass fraction of Cr_2O_3 in solid concrete)	= 0.0
fco_2	(Mass fraction of CO_2 in solid concrete)	= 0.2052
fh2Oe	(Mass fraction of evaporate water)	= 0.0
fh2Ob	(Mass fraction of chemically bound water)	= .0752
	Sum	= .847

Comments: Our concrete is composed of materials that are not listed in here or some materials that we don't have any information about. For now we add the mass fraction of all other materials to SiO_2; this makes our numbers almost consistent with Peach Bottom.