ENCLOSURE 4

Palo Verde Nuclear Generating Station
Spent Fuel Pool Boron Dilution Analysis (13-NS-C44)

Palo Verde Nuclear Generating Station

SPENT FUEL POOL BORON DILUTION ANALYSIS

Executive Summary

It is concluded that there is no source of water within the fuel building that could reduce the boron concentration of the spent fuel pool from the value of 2150 ppm (Technical Specification LCO 3.7.15) to 900 ppm (Ref. 79). A fire in the fuel building at elevation 140-ft. is the limiting event for boron dilution event and it bounds all normal, seismic and pipe break scenarios. This event would result in a final boron concentration in the spent fuel pool of 1,900 ppm (twice the minimum requirement K_{eff}).

It is possible for local dilution to exceed the minimum required value during an uncontrolled addition of non-borated water. This analysis assumes thorough mixing of all non-borated water added to the spent fuel pool. It is unlikely, with cooling flow and convection from the spent fuel decay heat, that thorough mixing would not occur. However, if mixing were not adequate, it would be conceivable that a localized pocket of non-borated water could form somewhere in the spent fuel pool. This possibility is addressed by the criticality calculation, which shows that the spent fuel rack K_{eff} will be, less than 1.0 with the spent fuel pool filled with non-borated water. Thus, even if a pocket of non-borated water formed in the spent fuel pool, K_{eff} will remain less than K_{eff} of 1.0 at anywhere in the pool.

The original design of the spent fuel pool cooling, clean up and supply systems also was reviewed and it had been concluded that the current design is capable of providing;

1. Sufficient borated makeup source to replenish boron during all events including loss off site power scenario,

2. Sufficient interlocks, instrumentation and isolation valves are provided in the design to make it possible for operators to mitigate any boron-dilution event.

3. Spent fuel pool cooling system as designed has ample heat removal capacity to maintain bulk temperature of the pool water to less than 145°F under all condition hypothesized by the Standard Review Plan (NUREG 800) with 1205 fuel assemblies in the pool. The maximum bulk temperature in pool will not exceed 167°F during a loss off site power scenario where operator action is required for restart of pool cooling pumps.

Additional design margin is provided by the Technical Requirement Manual (TRM) sections 3.1.104 and 3.1.105, which requires that the boron concentration in the fuel pool to be maintained to at least 4000 ppm (RCS back up borated water source).

In conclusion, the design of spent fuel pool / clean up and its auxiliary systems and fuel building have ample design margin to support proposed change in configuration of storage racks and additional decay heat loads.

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1.0 INTRODUCTION

The spent fuel pool configuration has been evaluated for possibility of boron dilution due to operational or accidental event and additional decay heat up to 1,205 fuel assemblies. The boron dilution analysis includes an evaluation of the following plant specific features:

- Dilution Sources
- Boration Sources
- Piping
- Instrumentation to mitigate a possible event.
- Administrative Procedures
- Boron dilution events

The boron dilution analysis was completed to ensure that the PVNGS design has sufficient margin to detect and mitigate the dilution before the spent fuel rack criticality analysis 0.95 K_{eff} design basis is exceeded.

The spent fuel pool cooling system has been design for total of 1205 spent fuel assemblies per the original design bases. A review of applicable design documents was performed to insure that all bounding accident and plant conditions are included in the design bases.

2.0 SUMMARY

It is concluded that there is no source of water within the fuel building that could reduce the boron concentration of the spent fuel pool from value of 2,150 ppm (Technical Specification LCO 3.7.15) to 900 ppm. A fire in the fuel building at elevation 140-ft. is the limiting event for boron dilution event and it bounds all normal, seismic and pipe break scenarios. As result of this event and conservative assumption used in analysis, 30,000 gallons of water could be discharged into the pool which could result in final pool concentration 1,900 ppm (twice the maximum required by criticality analysis). It is possible during introduction of non-borated water, for the local dilution to exceed the minimum required value. This analysis assumes thorough mixing of all the non-borated water added to the spent fuel pool. It is unlikely, with cooling flow and convection from the spent fuel decay heat; that thorough mixing would not occur. However, if mixing were not adequate, it would be conceivable that a localized pocket of non-borated water could form somewhere in the spent fuel pool. The criticality calculation, which shows that the spent fuel rack K_{eff} will be less than 1.0, addresses this possibility with the spent fuel pool filled with non-borated water. Thus, even if a pocket of non-borated water formed in the spent fuel pool, K_{eff} would not be expected to exceed 1.0 anywhere in the pool.

The original design of the spent fuel pool cooling, clean up and supply systems also was reviewed and it has been concluded that the current design is capable of providing;

1. Sufficient borated makeup source to replenish boron during all events including loss off site power scenario.

2. Sufficient interlocks, instrumentation and isolation valves are provided in the design to make it possible for operators to mitigate a boron-dilution event.

3. Spent fuel pool cooling system as designed has ample heat removal capacity to maintain bulk temperature of the pool water to less than 145°F under all condition hypothesized by the Standard Review Plan NUREG 800 with 1,205 fuel assemblies in the pool. The maximum bulk temperature in pool will not exceed 167°F during a loss off site power scenario where operator action is required for re-start of pool cooling pumps.

Additional margin is provided in the design since per Technical Requirement Manual (TRM) sections 3.1.104 and 3.1.105 requires the boron concentration in the fuel pool to be maintain to at least 4,000 ppm (RCS back up borated water source).

In conclusion, the design of spent fuel pool / clean up and its auxiliary systems and fuel building have ample design margin to support proposed change in configuration of storage racks storage capacity and additional decay heat load.

3.0 SYSTEM FEATURES

This section provides background information on the spent fuel pool and its related systems and features. A simplified P&ID diagram of the spent fuel pool and a general arrangement of related systems is provided as Figure 1 through 3.

3.1 Spent Fuel Pool

The spent fuel pool is a stainless steel lined, concrete walled pool that is an integral part of the fuel building. The spent fuel pool Seismic Category I physical boundary is defined as the outer gate (welded to the pool liner) located between the cask loading pit and the cask wash down area and the quick closure device on the containment side of the transfer tube in combination with the spent fuel pool liner and drain valves. During refueling operation, the physical boundary of the spent fuel pool is expanded and it includes the refueling pool and upper guide structure (UGS) pit (Seismic Category I) located in containment. Total capacity of the spent fuel pool during normal power operation is 340,000 gal (nominal). During refueling the total capacity of all pools is estimated to be 880,000 gal (nominal). The gate seals between the spent fuel pool and other pools are designed as NQR Seismic Category IX and the main function of these gates is to provide operational flexibility to move or relocate fuel within the spent fuel pool area (Ref. 1).

3.2 Spent Fuel Pool Storage Racks

The spent fuel pool storage racks are made up of individual modules seismically qualified. A module is an array of fuel storage cells. The storage racks are comprised of 17 modules: twelve 8 by 9 and four 8 by 12, and one 9 by 9 array with total maximum storage provided for up to 1,329 fuel assemblies. The storage racks are stainless steel honeycomb structures with rectangular fuel storage cells. The stainless steel construction of the racks is compatible with fuel assembly materials and the spent fuel borated water environment. The fuel assembly spacing of a nominal 9.5 inches center-to-center distance between adjacent storage cell locations are minimum values after allowances are

made for rack fabrication tolerances and predicted deflections resulting from a safe shutdown earthquake (SSE).

3.3 Fuel Pool Cooling and Decay Heat

3.3.1 Natural Convection, Water Circulation Within the Spent Fuel Pool

The PVNGS spent fuel pool storage racks are designed to ensure that fuel-cladding temperatures remain well below 650°F in an event of loss of pool cooling.

3.3.2 Forced Cooling Spent Fuel Pool and Shut Down Cooling Systems

As shown in figure 3, the spent fuel pool cooling system consists of two spent fuel pool cooling pumps, each powered from the Class 1E electrical system, and two fuel pool heat exchangers. The fuel pool cooling pumps have a common suction header and return header. Spent fuel pool water is circulated by the fuel pool pumps through the fuel pool heat exchangers, where it is cooled by the nuclear cooling water system or by the essential cooling water system and ultimate heat sink. Refer to table 9.2-1 of the UFSAR for system parameters.

The spent fuel pool cooling portion of the PC System is designed to be operated with the two cooling trains in parallel or each one individually. Normally, the spent fuel pool cooling system maintains the spent fuel pool bulk temperature below 125°F with both trains working or 145°F with one train operational (at maximum 1205 fuel assembly stored in the pool). Plant safety features include four provisions for cooling discharged fuel during any abnormal events;

- Spent fuel pool cooling train A
- Spent fuel pool cooling train B
- Shutdown cooling system train A (LPSI or Containment spray pumps)
- Shutdown cooling system train B (LPSI or Containment spray pumps)

The pool cooling system is cooled by the nuclear cooling system during normal operation. If the nuclear cooling water system is lost, essential cooling water system is aligned. The ECW system heat exchangers are serviced by the ultimate heat sink (spray ponds). Each of the two trains of the cooling system consists of a pump, a heat exchanger, valves, piping and instrumentation. All pumps take suction from the fuel pool at an inlet located below the pool water level (centerline elevation of 131 ft.), transfer the pool water through a heat exchanger and return it back into the pool through an outlet located below and a large distance away from the cooling system inlet (see figure 2). The return line is designed to prevent siphoning. The system design does not incorporate redundant active components except for the spent fuel pool pump and heat exchanger and associated valves. Alternate cooling capability can be made available under anticipated malfunctions or failures using shut down cooling

system. System piping is so arranged that failure of any pipeline does not drain the spent fuel pool below the top of the stored spent fuel assemblies.

3.3.3 Decay Heat

All spent fuel decay heat loads are calculated in Calculation 13-NC-RC-200 in accordance with the methodology established in Branch Technical Position 9-2. The maximum duration of the fuel cycle is taken as 24 months. The plant capacity factor is assumed at 90% with a plant power level of 3,954 MWt. The spent fuel pool has a design storage capacity to accommodate 1,329 fuel assemblies. The spent fuel pool cooling system, in conjunction with the shutdown cooling system provides adequate cooling for up to 1,205 fuel assemblies. The following heat load definitions are applicable to the pool cooling portion of the PC System.

1. "Normal Power Operation" Spent Fuel Pool Decay Heat Load - The "normal power operation" spent fuel pool decay heat load is based upon the spent fuel pool decay heat load at the end of a refueling outage when the unit resumes Mode 1 power operation. The design basis value for this heat load is determined in Calculation 13-NC-RC-200 to be 11.12×10^6 BTU/hr. This value assumes a partial core off-load (maximum of 120 assemblies) for a refueling duration of at least 34 days. In addition it is assumed that there are 964 assemblies that were offloaded during each of the previous 12 refuelings, each following a nominal 24-month fuel cycle. The spent fuel pool decay heat load is predicted to drop below the design basis decay heat load of 11.12×10^6 BTU/hr after 15 days following the reactor trip.

   Two trains of the spent fuel pool cooling system are available. One or two spent fuel pool cooling pumps would be in operation to keep the pool temperature below 125°F. In the event of failure of one train of spent fuel pool cooling, a single train is sufficient to maintain the spent fuel pool temperature below 145°F. The heat removal capability of the spent fuel pool cooling system under these conditions is 17.5×10^6 Btu/hr/train. These scenarios include accident conditions and full core offload outages. Depending on the scenario the spent fuel pool decay heat load could vary from 45.4E+6 to 9.68E+6 BTU/hr. One train of spent fuel pool cooling system (1 train of PC / EW cooled) augmented with one train of shut down cooling is capable of removing the maximum heat load under any condition (Table One).

   As actual refueling scenarios vary from outage to outage, the number of spent fuel assemblies actually transferred to the spent fuel pool may be greater or less than the number assumed in calculation 13-NC-RC-200. Additionally, the decay heat load per assembly will vary somewhat as a result of the actual fuel assembly design and irradiation period. As a result, administrative controls are in place to ensure that the actual heat load in the spent fuel pool does not exceed the design basis heat loads.

2. Emergency plant conditions - The spent fuel pool cooling system would be available within 8 hours from the initiating event. The system would be manually aligned with the essential cooling water system and ultimate heat sink system (this scenario is applicable to any event described in Chapter 15, which would result in loss of offsite power). During this scenario, spent fuel pool temperature would be limited to 167°F. The heat removal capability of one train of the spent fuel cooling system under these conditions is 11.12E+6 Btu/hr, and the maximum calculated decay heat generated in the spent fuel pool is 9.68E+6 Btu/hr. (Refer to UFSAR sections 9.1.3.1.1 and 9.1.3.2.1.1 for design bases and system descriptions.)

3. Scheduled normal refueling condition (includes full core offload reload, 100 hours after shutdown) - The spent fuel pool cooling system is normally cooled by one or two trains of the spent fuel pool cooling system and nuclear cooling water system. The shutdown cooling system (LPSI or containment spray pumps) can also be used to augment spent fuel pool cooling as needed. The spent fuel pool temperature during this mode of operation is maintained below 125°F. The heat removal capability of the spent fuel cooling system and shutdown cooling system under these condition is 77E+6 Btu/hr, and the maximum calculated decay heat generated in the spent fuel pool is 45.4E+6 Btu/hr. (Refer to UFSAR sections 9.1.3.1.1 and 9.1.3.2.1.1 for design bases and system descriptions.)

4. Emergency condition during a scheduled normal refueling (loss of offsite power and mechanical single failure) - One train of the spent fuel pool cooling system augmented by one train of the shutdown cooling system (LPSI or containment spray pump). These systems are cooled by essential cooling water in conjunction with the ultimate heat sink (spray pond). During this scenario, the spent fuel pool temperature would be limited to 145°F. The heat removal capability of one train of the spent fuel cooling system and one train of the shutdown cooling system under these conditions is 46.1E+6 Btu/hr, and the maximum calculated decay heat generated in the spent fuel pool is 45.4E+6 Btu/hr. (Refer to UFSAR Sections 9.1.3.1.1 and 9.1.3.2.1.1 for design bases and system descriptions.)

5. Emergency condition during fuel transition mode (loss of off-site power and a mechanical single failure) - During this mode of operation, when the core is being off loaded or re-loaded, one train of the fuel pool cooling system could be augmented by one train of the shutdown cooling (LPSI or containment spray pump) and associated auxiliaries. The shutdown cooling train in service is aligned such that it would provide cooling to both the reactor core and spent fuel pool. The maximum pool temperature during this condition would be limited to 145°F. The heat removal capability of one train of the spent fuel cooling system and one train of the shutdown cooling system under these

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conditions is 45.4E+6 Btu/hr, and the maximum calculated decay heat generated in the spent fuel pool is 45.4E+6 Btu/hr. (Refer to UFSAR Sections 9.1.3.1.1 and 9.1.3.2.1.1 for design bases and system descriptions.)

6. Emergency core off load condition - One train of fuel pool cooling and one train of shutdown cooling can maintain and limit the maximum spent fuel pool temperature to less than 125°F. Fuel pool decay heat for this event is a full core off loaded 100 hours after shutdown plus 1/3 core offload 90 days before the full core offload plus 884 assemblies from the 11 previous annual refuelings. This same temperature limit will apply even with a mechanical single failure, as there is a spare train available for each system. The heat removal capability of one train of the spent fuel cooling system and one train of the shutdown cooling system under these conditions is 59.5E+6 Btu/hr, and the maximum calculated decay heat generated in the spent fuel pool is 45.4E+6 Btu/hr. (Refer to UFSAR Sections 9.1.3.1.1 and 9.1.3.2.1.1 for design bases and system descriptions.)

Table 1 - Pool Cooling / Shutdown Cooling Capacity and SFP Decay Heat

Parameter		Value
Minimum Heat Removal Capability, Btu/hr		
1 -	Normal plant condition two pool cooling/two NC cooling	35E+6
2 -	Emergency plant condition one pool cooling/one EW cooling	11.12E+6
3 -	Scheduled Normal refueling plant condition	
	Pool cooling (2 train)/two NC cooling	35E+6
	Shutdown cooling (1 train)/one EW cooling	42E+6
	Total	77E+6
4 -	Emergency plant condition during a scheduled refueling	
	Pool cooling (1 train)/EW cooling	16.5E+6
	Shutdown cooling (1 train)/EW cooling	29.6E+6
	Total	46.1E+6
5 -	Emergency plant condition during fuel transition mode (off load/reload)	
	Pool cooling (1 train)/EW cooling	15.3E+6
	Shutdown cooling (1 train)/EW cooling	30.1E+6
	Total	45.4E+6
6 -	Emergency core off load during an unscheduled outage	
	Pool cooling (1 train)/NC cooling	17.5E+6
	Shutdown cooling (1 train)/EW cooling	42.0E+6
	Total	59.5E+6
Spent Fuel Decay Heat (a), Btu/hr Plant Condition		
1 -	Design heat load power operation Partial core (maximum of 120 assemblies) offload for refueling duration of longer than 34 days and 964 assemblies offloaded from the previous annual refuelings.	11.12E+6
2 -	Design heat load during emergency plant condition (accident) 120 fuel assemblies offloaded 45 days prior to accident plus 964 assemblies from the previous annual refuelings.	9.68E+6

Table 1 - Pool Cooling / Shutdown Cooling Capacity and SFP Decay Heat, Continued

Parameter		Value
Spent Fuel Decay Heat (a), Btu/hr Plant Condition (continued)		
3, 4, and 5 -	Design normal/emergency refueling heat load Full core offload 100 hours after shutdown plus 964 assemblies from the previous annual refuelings.	45.4E+6
6 -	Design Maximum heat load during emergency core offload Full core offload 100 hours after shutdown plus 1/3 core offload 90 days before the full core offload plus 884 assemblies from the previous annual refuelings.	45.4E+6

3.4 Spent Fuel Pool Clean up System

The spent fuel pool clean up system consists of two trains, each having a strainer, a pump, a filter, and an ion exchanger. Either one or both trains may be aligned to clean the water in the spent fuel pool or the refueling water tank continuously (if required), or the refueling pool intermittently. The spent fuel pool clean up system is aligned with the pool drain system only for a short duration of time during the refueling for fill and drain / clean up operations. The system is under procedural control during these evolutions. The clean up loops are normally run from a local panel. It is possible to operate each loop independently and with either the ion exchanger or the filter bypassed by means of manually operated valves.

3.5 Pool Drain System and Leak Detection

There are no drains within the spent fuel pool. However, drains are provided in the cask-loading pit, fuel transfer canal, refueling pool and UGS pit. Drain system provides the plant with capability of fill and drain operation during refueling or fuel transfer and fuel removal. The design of spent fuel pool includes a leak detection system. The leak detection system is isolated by design and it is only opened during administrative walk downs to insure that integrity of liner is maintained.

3.6 Gates, Permanent and Pneumatic Seals

All gates in the fuel building are designed with pneumatic seals (single bladder, NQR) with a redundant / manual air supply system. The primary source of air for these seals is a plant instrument air system (IA). The back up source is provided by bottled compressed gas located in the fuel building. During refueling operation, PVNGS cavity seal is in place. This is a permanent passive seal. Also during refueling the steam generators dams (NQR) could be utilized for steam generator inspections. These dams are designed with redundant gas supply system and seals (double bladder). Similar to spent fuel pool gates, the primary source of gas for these seals is the plant instrument air system (IA). The back

up source is provided by bottled compressed gas located in the containment building. Table 2 summarizes the location and geometry of pool seals.

Table 2 - Summary of Pool Gates and Seals

Gate Name	Location	From / To	Required Mode of Operation	Type of Seal / Number	Minimum Elevation
Fuel transfer canal	Fuel building	Transfer canal to fuel pool	1- 6 and de-fuel	Pneumatic seals / (single bladder)	113' – 10"
Cask loading pit	Fuel building	Spent fuel pool to cask loading pit	1- 6 and de-fuel	Pneumatic seals / (single bladder)	113' -10"
Cask wash down	Fuel building	Cask loading pit to cask wash down area	NA	Welded in place	124'
Cavity refueling seal	Contain-ment building	Refueling pool to reactor cavity	5- 6 and Refueling / de-fuel	Passive – permanent seal	114'

3.7 Normal System Leakage

The normal spent fuel pool leakage is divided to 1) system leakage, 2) evaporation and 3) the pool boundary leakage. The normal system leakage due to valve stem flanges, piping and pump mechanical seals is limited to 5 gpm and it is controlled by administrative procedure. Pool evaporation losses are estimated to be no more than 2 gpm at pool bulk temperature of 145 °F. The boundary leakage is a combination of leakage through PCN-V118 and the liner of spent fuel / refueling pool. Allowable leakage through this pathway is administratively controlled to 30 gpm. During normal plant operations the losses from the system are dominated by evaporation loss and total leakage is significantly lower than design values stated .

3.8 Location of Siphon Breaker Location Within PC Systems

Anti Siphon holes are provided in all piping that penetrates the spent fuel boundary above the fuel rack elevation to eliminate the possibility of loss of coolant below minimum required water level in a seismic event. The minimum elevation for this design feature has been selected to insure that at no time would pool cooling be lost due to siphon effects.

Table 3 - Designs Make up Sources to the Spent Fuel Pool

Piping Description	System	Isolation Valve Tag Number	Classification	Anti Siphon Hole Elevations
Fuel pool clean up suction	Pool clean up (PC)	V080	NQR / Cat III	133' 0"
Boric acid makeup pumps suction / discharge line	Chemical and volume control (CVCS)	V144B	Q / Cat I	133' 6"
Fuel pool cooling discharge	Pool cooling (PC)	V026 / V027	Q / Cat I	132' 6"
Condensate Pump makeup line	Condensate transfer system (CT)	V019	NQR / Cat II	135'
Radwaste monitor tank makeup	Liquid radwast (LR)	V250	NQR / Cat III	135'

3.9 Radiation Shielding

The fuel stored in spent fuel pool racks within the pool is normally shielded a approximately 23 ft. of water. Direct dose rate at surface of pool (i.e. refueling machine) from a fully filled pool is less than 2.5 mrem. This provides an unlimited access to operation personnel.

3.10 Equipment Qualification of Safety Instrumentation / Components

The spent fuel pool does not house any safety related electrical / electronic components. An increase in number of fuel assemblies stored within the pool has no impact on equipment qualification program. As stated in section 3.9, the dose rate at boundary of spent fuel pool will remain below 2.5 mrem and within the original radiation design dose rates in the fuel building general area. Therefore, there is no impact to equipment / components housed within the fuel building. All gate seals within the spent fuel boundary are NQR, although not required for nuclear safety, seals material degradation due to the fuel assembly configuration were reviewed. It is concluded that original design exposure due to gamma radiation would remain bounding. The effects of beta and alpha radiation have been neglected due to shielding provided by water, the spent fuel racks and concrete. Neutron damage is minimal due to large water / concrete shielding between spent fuel racks and gates.

3.11 Quality Classification

The spent fuel pool cooling components are a part of the quality related (Q), Seismic Category I, categorized as Quality Group C (regulatory guide 1.26) system. The pool clean up system has a minimal quality assurance requirement and has been classified as NQR-), Seismic Category III.

All pool drain systems in the transfer canal, cask loading pit, refueling pool and UGS pit, up to and including the first manual isolation valves are designed to withstand SSE and

are quality related. These valves are normally in closed position and they are controlled by administrative "locked valve" procedure.

4.0 SOURCES OF MAKE UP WATER TO THE SPENT FUEL POOL AND THE REFUELING POOL

The design requirements for the spent fuel pool makeup system have been changed as a result of the boron credit in the spent fuel pool. The primary makeup source to the spent fuel pool has been changed from a non-borated source (liquid radwaste system / condensate transfer system) to a borated source (boric acid makeup pumps). All non-borated sources to spent fuel and refueling pools have been administratively isolated. Additional requirements will be imposed to administratively estimate and verify the boron content of the spent fuel pool by a chemistry technician each time non-borated sources are used to add inventory to the pool.

During normal plant operation, addition of non-borated water is allowed by design since evaporation losses would result in continuous buildup of the boron concentration. Liquid Radwaste system monitor tanks and the Condensate Storage Tank (CST) are used for replenishing the pool inventory. To minimize possibility of local dilution, chemistry procedures will be revised to include direction to add inventory to the pool at a preset maximum flow rate.

4.1 Borated Make up Sources (Seismic / Quality)

4.1.1 Refueling Water Tank

The Refueling Water Tank (CHE-T01) stores borated water for use in the Reactor Coolant System. The RWT is sized to allow total boric acid recycle, to allow back-to-back cold shutdowns to 5% subcritical and subsequent startups at 90% of core life. The tank capacity also includes inventory for filling the Refueling Pool, and includes approximately 476,000 gallons of borated water reserved for Engineered Safety Feature (ESF) pump operation. This reserved ESF volume provides the minimum operating volume for the Safety Injection system functions. The RWT has a nominal usable volume of 720,000 gallons and is maintained at a boron concentration in the range of 4,000-4,400 ppm (Ref. 3). The tank is designed to Seismic Category I requirements (Ref. 2).

The minimum Technical Specification level for the RWT corresponds to a tank level of 80%. This tank level equates to a plant elevation of 142'-2" and provides 600,000 usable gallons of borated water inventory. The usable volume includes the tank inventory above the level of the SI system suction strainer/vortex breaker. The tank is equipped with a Low-Level alarm, which actuates at a tank level of 87%. This Low-Level alarm represents a plant elevation of 146'-5" and provides for a usable borated water inventory of about 652,000 gallons (Ref. 4). Based on these values, there are approximately 52,000 gallons of RWT inventory between the Low-Level alarm setpoint and the minimum Technical Specification level. This 52,000 gallons of RWT inventory would typically be available for makeup to the Spent Fuel Pool.

4.1.2 Borated Makeup Sources for the SFP

The Chemical Volume and Control System (CVCS) was originally scoped to provide a borated makeup water supply to the Spent Fuel Pool (SFP). The CVCS was designed to supply the SFP makeup through the Pool Cooling and Purification System (PC). A dedicated piping run connects the CVCS system directly with the SFP to provide this makeup flow path. The Boric Acid Makeup Pumps (BAMPs) were the components originally designated to provide this borated water supply to the SFP. The Combustion Engineering CVCS Interface Requirements for the BAMPs stipulated that the pumps would supply the SFP with a makeup flow rate of 165 gpm at a delivery pressure of 130 PSIG (Ref. 5). Figure 4 contains a simplified diagram of the SFP makeup system flow paths.

4.1.3 Spent Fuel Pool Makeup Utilizing the BAMPs

The Refueling Water Tank (RWT) is the normal source of borated makeup water for the Spent Fuel Pool. The RWT inventory is maintained at a boron concentration of 4,000 to 4,400 ppm in accordance with Technical Specification requirements (Ref. 6, 7 & 8). The BAMPs (CHN-P02A & B) are utilized to transfer this borated makeup water from the RWT to the SFP. The centerline of the suction piping for the BAMPs enters the RWT at a plant elevation of approximately 134 feet (Ref. 9). This elevation corresponds to a storage tank level of 66% (Ref. 14). There is a hard-wired RWT Low-Level cutout for the BAMPs at a tank level of 73% (Ref. 16). As such, the RWT must be at a level of 73% or greater to utilize the BAMP flow path for makeup to the SFP. This 73% RWT level corresponds to a plant elevation of 141'-2" (Ref. 10).

Utilizing the BAMPs to makeup to the SFP from the RWT requires Operations to manually align the system components (Ref. 11). In this alignment, valve PCN-V215 is opened to connect the BAMP discharge piping to the SPF. This valve is located in the 100-foot elevation of the Fuel Building. A BAMP is then started and the pump discharge pressure is manually throttled to 130 psig at valve CHN-V753. Valve CHN-V753 is located in the 70-foot elevation of the Auxiliary Building. The makeup flow rate to the SFP is approximately 165 gpm under these throttle conditions.

The RWT, BAMPs and the piping and valves associated with this borated water makeup path are designed to Seismic Category I requirements (Ref. 12). Hydraulic calculations have been performed for this makeup flow path from the RWT, through the BAMPs to the SFP (Ref. 13). The results of these analyses indicate that this is a viable makeup flow path. Makeup flow rates on the order of 165 gpm are readily obtainable with single pump operation.

4.1.4 SFP Makeup via Gravity Flow from RWT

The RWT can be manually aligned by Operations to provide borated makeup water to the SFP via gravity feed flow path (Ref. 14). This alternate flow path

can be utilized to makeup to the SFP from the RWT should the BAMPs be unavailable. This flow path uses the normal RWT to BAMP suction piping run. For this alignment, valve CHN-V144 is manually opened to bypass the flow from the RWT around the BAMP suction piping. This valve is located in the 70-foot elevation of the Auxiliary Building. Valve PCN-V215 is then opened to route the gravity feed flow from the RWT to the SFP. The RWT, along with the piping and valves associated with this borated water makeup path, are designed to Seismic Category I requirements (Quality Classification Q1C) (Ref. 12 & 15).

This makeup flow path relies on gravity to provide the motive force. The fluid level in the RWT must be at a higher elevation than the fluid level in the SFP for makeup flow to occur. A hydraulic analysis performed for this makeup flow path indicates that an initial flow rate on the order of **68 gpm** can be obtained with this system alignment (Ref. 13). This initial flow rate assumes a minimum RWT level of 87%, which corresponds to the Low-Level alarm setpoint for the tank (Ref. 16). The gravity feed flow rate from the RWT will decrease as the level in the tank is reduced due to the makeup flow to the SFP. The analysis performed for this alignment indicates that a gravity flow rate on the order of **41 gpm** will be delivered to the SFP for an RWT level of 80%. This 80% level is the minimum Technical Specification level for Operability of the RWT (Mode 1) (Ref. 6 & 7).

Use of this flow path is currently limited by procedure to RWT levels above a value of 73% (Ref. 14). Administrative controls currently in place ensure that Operations complies with the Technical Specification Limiting Conditions for Operations (LCOs) for the RWT. The operability of the RWT is impacted (Mode 1) for tank levels below the minimum Technical Specification value of 80%.

4.1.5 SFP Boration Utilizing the Boric Acid Batch Tank

The Boric Acid Batch Tank (BABT), in conjunction with the BAMPs, can be utilized to borate the contents of the SFP (Ref. 17). The BAMPs are manually aligned by Operations to take suction from the SFP through valve PCN-V215. The BAMP discharge flow rate is throttled to **125 gpm** through controller CHN-FIC-210Y and directed to the Boric Acid Batching Eductor (CHN-J01). A concentrated solution of approximately 12% Boric Acid by weight is initially prepared in the BABT per the operating procedure requirements (Ref. 17). The contents of the BABT, a volume of approximately 500 gallons, are then educed into the BAMP discharge flow. The borated flow stream is directed back to the SFP through manual valves CHN-V024 and PCN-V080. Valve CHN-V024 is located at the plant south side of the RWT, and valve PCN-V080 is located at the 120-foot elevation of the Fuel Building. A simplified diagram of this boration flow path is shown in Figure 5.

The impact of the batching operations on the SFP boron concentration can be calculated utilizing the formulas contained in the BABT operating procedure

(Ref. 17). For an initial SFP Boron concentration of 4,200 ppm, a single BABT addition will raise the SFP Boron concentration by approximately 26 ppm. This assumes an initial SFP volume of 320,000 gallons, which excludes the volumes of the Cask Loading Pit (CLP) and the Fuel Transfer Canal (FTC) (Ref. 76). This determination also assumes an initial BABT volume of 489 gallons and a solution concentration of 12% Boric Acid by weight. Additional batching operations can be made as required to increase the SFP Boron concentration to the desired range. Existing operational procedures require the Chemistry Section to perform an analysis for Boron concentration in the SFP following any batching operations.

A portion of the piping and components associated with this boration flow path are designed to non-Seismic Category I requirements (Ref. 12 & 15). These components include the Boric Acid Batching Eductor and portions of the return piping to the SFP. In addition, if the level in the RWT is below a value of 73%, the RWT Low-Level cutout for the BAMPs will normally prevent the pumps from operating. Under these circumstances, Electrical Maintenance must be contacted to physically bypass the RWT Low-Level cutout in order to utilize this boration flow path. The BABT operating procedure (Ref. 17) includes an appendix with steps for Electrical Maintenance to bypass this RWT Low Level cutout for the BAMPs.

Based on operational experience, approximately 2 ½ hours are required to prepare a batch of the ~12% Boric Acid solution for addition to the SFP. This process involves heating the water in the BABT and mixing in the bulk boric acid reagent. Up to an additional hour is required to transfer the contents of the BABT to the SFP depending on the flow from the BABT to the eductor (Ref. 17).

4.1.6 Low-Pressure Safety Injection / Shutdown Cooling Pumps

The low-pressure safety injection (shutdown cooling) and the containment spray pumps can provide borated water to the spent fuel pool from the refueling tank, if required. A simplified diagram of these makeup flow paths is shown in figure 6. Although this is not the primary function of these pumps, the line up and, procedural requirement (40AO-9ZZ23 and Ref. 14) are in place to provide the spent fuel pool with borated water source in a case of loss of the primary makeup source or degraded spent fuel pool condition. These single stage centrifugal pumps have a rated flow of 4,300 gpm. The operators can control the makeup rate by throttling valves SIA-HV-306 in train A or, SIB-HV-307 in train B. The make up flow rate for this source can vary from 150 gpm to 5,000 gpm (the minimum flow rate of 100 gpm can only be utilized for pump operating periods of one hour or less). The source of water for the low-pressure safety injection (shutdown cooling) and the containment spray pumps is the refueling water tank. This tank is designed and it is maintained to boron concentration of within 4,000 – 4,400 ppm (Technical Specification SR 3.5.5.3). Minimum makeup capacity of RWT is 52,000 gal (volume between low alarm @ 87.8% and Technical Specification minimum requirement @ 80%,

Figure 3.5.5-1, TS 3.5.5). Use of this system for make up is restricted by limitations of the Technical Specification and the design requirements during all modes of operation.

4.2 Non-Borated Makeup Sources

4.2.1 Condensate Transfer Pump / Condensate Storage Tank

The condensate transfer system is directly connected to the spent fuel pool. Using the direct connection, the contents of the condensate storage tanks can be transferred via the condensate transfer pumps directly to the spent fuel pool. The direct connection is normally isolated from the condensate transfer system by a closed manual valve (CT-HV-019). The direct connection is used as the normal water supply to the spent fuel pool and it is a source of make up water in case of replenishment due to pool evaporation.

The condensate transfer system consists of one-tank and two condensate transfer pumps per each unit. The condensate storage tank (CST) is located in the north yard area and is a concrete tank lined with ¼" stainless steel, inner diameter of 46.5 ft and height of 53 ft. The tank is designed as a seismic category I. It has a nominal capacity of 550,000 (Ref. 19, 24) gallons of demineralized water. However, only 528,000 (Ref. 20) gallons inventory are above the suction line of the condensate transfer pumps. The condensate transfer pumps (two) are housed in seismic category I structure and they are quality related. The make up to the spent fuel pool can be provide a approximately at a flow rate of 100 (maximum capacity of pump is 130 gpm) of gpm / pump (Ref. 21, 25). The condensate transfer system piping is Q / seismic category I, with exception of piping branch from discharge of CT pump to the spent fuel pool. This portion system is designed to ASME B31.1 code and seismic category II. Due to differential height between the CST (maximum water elevation 158 ft.) and the spent fuel pool (minimum elevation 131ft.) (Ref. 22,23), makeup can be established by gravity flow. The addition of CST water to the spent fuel pool is administratively controlled by procedure 4XAO-9ZZ23.

4.2.2 Liquid Radwaste Recycle Monitor Pumps / Radwaste Recycle Monitor Tanks

The recycle monitor pump delivers water from the recycle monitor tanks (RMT) to the spent fuel pool if water is acceptable for reuse. The system is designed to the requirements of the Regulatory Guide 1.143, and it is QAG / seismic category II. The supply system is comprised of two pumps and two tanks (Ref. 26). Housed in radwaste building, the tanks are located in the plant south coordinate at grade elevation. Each tank has 31,400-gallon capacity with outer diameter of 15 ft (Ref. 27). and height of 25 ft. The recycle monitor pumps are horizontal, centrifugal pumps with rate capacity of 150 gpm. No gravity flow is possible through this system since the elevation of the top of monitor tanks is

below the minimum spent fuel pool water level. The addition of RMT water to the spent fuel pool is administratively controlled by procedure 4XAO-9ZZ23.

4.3 SFP Makeup Utilizing the RWMT

The Reactor Water Makeup Tank (CHN-T02) is designed to store non-borated water for use in the Reactor Coolant System. The tank is sized to allow total recycle. The tank also provides for dilution for back-to-back cold shutdowns and subsequent startups at 90 percent of core life (Ref. 2). This component does not provide any safety related function nor is it required for safe shutdown of the plant. The RWMT is designed to Non-Seismic Category I requirements (Quality Classification R2I) (Ref. 71). The RMWT has a nominal storage capacity of 480,000 gallons. Two Reactor Water Makeup Pumps (RWMPs), (CHE-P03A & B) take their suction from the RWMT. These pumps are identical to the BAMPs and are designed to provide a nominal flow rate of 165 gpm at a discharge pressure of 130 psig (Ref. 2). The inventory level in the RWMT must be above 5 feet to utilize this makeup flow path.

The RWMT is manually aligned to the SFP through the RWMPs by Operations (Ref. 14). This makeup flow path utilizes the piping and valves associated with the Boric Acid Batching Eductor. Current procedural guidance directs that the discharge flow rate of the selected RWMP be throttled to 125 gpm through controller CHN-FIC-210Y. The non-borated makeup flow is directed to the SFP through valve PCN-V080. Valve PCN-V080 is located at the 120-foot elevation of the Fuel Building. Portions of the piping and components associated with this flow path are designed to non-Seismic Category I requirements (Ref. 12 & 15).

4.4 Reliability of SFP Makeup Sources

4.4.1 RWT via BAMPs

The electrical power for the BAMPs is supplied by the 480V Non-Class 1E Power System. Pump CHN-P02A is supplied by Motor Control Center (MCC) E-NHN-M13, which in turn is ultimately, supplied by Non-Class 1E bus E-NAN-S01 (Ref. 28). Pump CHN-P02B is supplied by Motor Control Center (MCC) E-NHN-M10, which in turn is eventually supplied by Non-Class 1E bus E-NAN-S02 (Ref. 29). These power sources, and subsequently the BAMPs, would not be available for use following a Loss of Off-site Power event.

As noted previously, the RWT storage tank, the BAMPs, the connecting piping, valves and components associated with this makeup flow path are designed to Seismic Category I requirements. Seismic Category I requirements are applied to those Structures, Systems and Components (SSC) that must remain functional during a Safe Shutdown Earthquake (SSE). Based on these considerations, this makeup flow path from the RWT to the SFP through the BAMPs would remain intact in the event of an SSE. However, as noted above, the power supply for the BAMPs is Non-Class 1E and is not designed to withstand a seismic event.

4.4.2 RWT via Gravity Flow

The gravity feed flow path from the RWT to the SFP does not require electrical power to function. There is one remotely operated valve in this flow path (CHE-HV-532). This component is an Air Operated Valve (AOV) which fails in the open position (Ref. 12). As noted previously, Operations must manually align two valves to initiate this SFP makeup flow path. The capabilities of this flow path are not affected by a Loss of Offsite Power.

As previously described, the RWT storage tank, the connecting piping, valves and components associated with this makeup flow path are designed to Seismic Category I requirements. Seismic Category I requirements are applied to those Structures, Systems and Components (SSC) that must remain functional during a Safe Shutdown Earthquake (SSE). Based on these considerations, this gravity feed flow path from the RWT to the SFP would be available in the event of an SSE.

4.4.3 RWT via Safety Injection System

The 4.16 kV Class 1E Power System provides electrical power for the Containment Spray and Low-Pressure Safety Injection pumps. The 4.16 kV Class 1E buses, E-PBA-S03 and E-PBB-S04 supply these SI system pumps, for the A and B Trains respectively (Ref. 30 & 31). The Class 1E Power System contains redundant standby Diesel Generator (DG) power sources. These DG sources automatically provide the power required for safe shutdown in the event of a loss of voltage on either of the Class 1E buses. The capabilities of this flow path to provide makeup from the RWT to the SFP are not affected by a Loss of Offsite Power event.

As described previously, the RWT storage tank, SI system pumps, connecting piping, valves and components associated with this makeup flow path are designed to Seismic Category I requirements. Seismic Category I requirements are applied to those Structures, Systems and Components (SSC) that must remain functional during a Safe Shutdown Earthquake (SSE). Based on these considerations, this makeup flow path, from the RWT to the SFP via the SI system pumps, would be available in the event of an SSE.

4.4.4 Refueling Water Makeup Tank

The electrical power for the RWMPs is supplied by the 480V Non-Class 1E Power System. Pump CHN-P03A is supplied by Motor Control Center (MCC) E-NHN-M15, which in turn is ultimately, supplied by Non-Class 1E bus E-NAN-S01 (Ref. 32). Pump CHN-P03B is supplied by Motor Control Center (MCC) E-NHN-M10, which in turn is eventually supplied by Non-Class 1E bus E-NAN-S02 (Ref. 29). These power sources, and subsequently the RWMPs, would not be available for establishing a makeup flow path to the SFP, following a Loss of Offsite Power (LOOP).

The RWMT, the RWMPs, along with the connecting piping, valves and components associated with this makeup flow path are designed to Non-Seismic Category I requirements (Ref. 71). Seismic Category I requirements are applied to those Structures, Systems and Components (SSC) that must remain functional during a Safe Shutdown Earthquake (SSE). Based on these considerations, this makeup flow path from the RWMT to the SFP through the RWMP was not designed to remain functional in the event of an SSE.

5.0 DILUTION SOURCES

In this section of the report, possibilities of dilution from all other sources of non-borated water are evaluated. In general, sources are not directly connected to the spent fuel pool and a procedural error or a failure of equipment is hypothesized.

5.1 Fire Protection System

In a case of fire in the spent fuel building, the local fire hose station (one Class III station) at elevation 140-ft. is the only source of potential non-borated water (4 inch piping). This station is capable of providing approximately 100 gpm of non-borated water under normal conditions. However, for purpose of this evaluation, a conservative maximum flow rate of 500 gpm will be used (Ref. 34). There are four other fire protection hose stations within the fuel building. These stations are below the spent fuel pool operating deck (elevation 140 ft.) and they will not be used for suppression of a fire at 140 elevation. Also the area west of cask loading pit (Fire Zones 27) (Ref. 35) has a wet pipe sprinkler system (at elevation 190' –6") (Ref. 36, 37). This zone is separated from the spent fuel pool by distance greater than equivalent of ½ nozzle sprayed diameter (8 ft.). The over spray from the wet sprinkler system would be minimal. However, for purpose of this evaluation, it is conservatively assumed that 50 percent of all adjacent nozzles to the cask loading pit flow would be released and mixed with pool inventory. The estimated non-borated flow from this source is 45 gpm (6 sprinklers @ 15 gpm/sprinkler).

The fire protection system is a quality augmented – non- seismic system. The fire protection lines in the fuel building are designed to the seismic category IX and they are categorized as moderate energy lines (system pressure of 175 psig and system temperature of less than 120°F). If any of these lines were to break, the flow rate from a through wall crack is estimated to be less than 85 gpm (Ref. 38). Assuming 60-min isolation due to the high level alarm generated by the increased fuel pool level in control room. The total discharge to the pool would be approximately 5,100 gal.

5.2 Demineralized Water System

The demineralized water system (DWS) does not connect directly to the spent fuel pool cooling system or the spent fuel pool. However, a number of one-inch connections are provided at elevation 144 ft. to facilitate the decontamination stations and utility outlets. This system is supplied by a main header at elevation 131 ft. The main header consists of a 3" stainless steel line that branches to 1½ inches prior to extending above elevation 140 ft. There are five decontamination /.utility stations above 144 ft. and each one is

equipped with a shut off valve and hose connection. Three of five decontamination stations are located in the adjacent area to the spent fuel pool. The other two decontamination stations outlets are located adjacent to the cask decontamination pit (floor elevation 113 ft), and approximately 20-ft. from the pool boundary (Ref. 39 – 41). The demineralized water system is a non-quality and non-seismic design. Lines adjacent to the spent fuel pool are designed to the seismic category IX and they are categorized as moderate energy lines (system pressure of ~130 psig and system temperature of less than 120°F). Each decontamination / utility station can provide a nominal flow of 50 gpm. Administrative procedures are in place (Ref. 42) to control addition of demineralized water to the spent fuel pool and other pits.

If any of these lines were to break, the flow rate from a through wall crack is estimated to be less than 23 gpm (Ref. 38). Assuming a 60-min isolation due to the high level alarm generated by the increased fuel pool level in the control room, the total discharge to the pool would be approximately 1,370 gal. If the break is not isolated by an operator action, it would result in overflowing the spent fuel pool boundary and the spillage would be released into the 100-ft elevation and then to the yard area outside the spent fuel building.

Pool clean up mixed bed ion exchangers (PCN D01A and D01B) are located at the auxiliary building elevation 120 ft. After isolation of an ion exchanger, the reactor make up water is used for sluicing resin to the solid radwaste system resin hold up tanks. To refill the ion exchangers vessels with resin, demineralized water is used. The vent and fill process can be performed using either spent fuel pool water or demineralized water (Ref. 42 through 44). The source of dilution for this process is minimal since the demineralizers are isolated from PC before this evolution. The only possible dilution event, which could accur, is if an ion exchanger is filled with demineralized water before it is put into the service. The maximum volume of non-borated water that can be injected to the spent fuel pool is approximately 825 gallons (volume of one vessel, no credit is taken for resin volume).

5.3 Domestic Water System

The Domestic Water system, DS, is a collection of supply, treatment, storage, and distribution subsystems, which are utilized to produce and distribute, filtered and chemically treated water to miscellaneous utility services. The domestic water headers in the fuel building consist of 2-inch main pipe. There are four (4) one inch copper branches for utility stations at elevation 140 ft. Each utility station is provided with a root valve that is normally in the close position (Ref. 45, 46). The domestic water system is a non-quality and non-seismic design. Lines adjacent to the spent fuel pool are designed to the seismic category IX and they are categorized as moderate energy lines (system pressure of ~ 65 – 80 psig and system temperature of less than 105 °F). Each utility station can provide a nominal flow of 50 gpm. At no time is this source of water used for make-up to the spent fuel pool. If any of these lines were to break, the flow rate from a through wall crack is estimated to be less than 4 gpm (Ref. 38). The total discharge to the pool would be small and the level increase will take significant time.

5.4 Nuclear Cooling Water (NCW) and Essential Cooling Water Systems (ECW)

The nuclear cooling water or essential cooling water system is the cooling medium for the spent fuel pool cooling system heat exchangers. There is no direct connection between the component cooling system and the spent fuel pool cooling system. If however, a leak were to develop in a heat exchanger that is in service, the connection would be made. In case of a leak the spent fuel pool water would be expected to leak into the cooling systems.

The spent fuel pool cooling system normally operates at slightly lower pressure than the nuclear cooling system or the essential cooling water system (~ 50 psig) (Ref. 47, 48 and 50). When both spent fuel pool cooling pumps are operating, and since the operating pressures of the two systems are very close, it is feasible for a spent fuel pool cooling system heat exchanger tube leak / rupture to result in cooling system water leakage into the spent fuel pool cooling system. It would be expected that the flow rate of any leakage of component cooling water into the spent fuel pool cooling system would be very low due to the small difference in operating pressures between the two systems. Assuming a 10 psid differential (cooling water system is 10 psi higher than pool cooling), the flow through a crack would be approximately 1 gpm and for tube rupture would be less than 85 gpm (Ref. 51, 52). Any loss of water from the component cooling system surge tanks would be automatically replaced with demineralized water, which could increase the amount of water available to dilute the pool (Ref. 53, 54). However, alarms and control room indications would alert the control room operators to any significant loss of water from the component cooling system during events such as heat exchanger tube rupture. In case of leaks, the duration of leakage would be extended since alarms will not be readily activated. The control room would be identifying the leakage by the frequency of level change in surge tanks or the increase in pool elevation. If a leak is very small and it is less than evaporation rate from the spent fuel pool, it may go undetected for the duration of the fuel cycle. However, this type of leakage would not have an impact on boron concentration since the quantity of boron the pool would remain unchanged.

5.5 De-Boration by Pool Clean up Ion-Exchangers

When the spent fuel pool demineralizer is first placed in service after being recharged with fresh resin, it will initially remove boron from the spent fuel pool. In the worst case, assuming pure anion resin, the demineralizer could remove a maximum of 7 to 10 ppm of boron from the spent fuel pool water before the resin would become saturated. The demineralizer normally utilizes a mixed bed of anion and cation resin, which would remove less boron before saturating. Because of the small amount of boron removed by the demineralizer, it is not considered a credible dilution source for the purposes of this evaluation.

5.6 Dilution Source and Flow Rate Summary

Based on the evaluation of potential spent fuel pool dilution sources summarized above, the following dilution sources where determined to be capable of providing a significant amount of non-borated water to the spent fuel pool.

Table 4 - Summary of Dilution Sources for Spent Fuel Pool

	Normal	Event Related	Pipe Break
Condensate transfer system	100 (gpm)	NA	NA
Liquid radwaste recycle monitor tank	150 (gpm)	NA	NA
Fire Protection Supply Lines	NA	45 - 500[1](gpm) (Fires)	85 (gpm)
Demineralized Water System			
- Decon / Utility station	NA	50 (gpm) (Operator error)	23 (gpm)
- Spent Fuel Pool Clean up System resin change out	850 Gal / change	NA	NA
Domestic water Utility Station	NA	50 (gpm) (Operator error)	4 (gpm)
Nuclear cooling / Essential cooling	1 (gpm)	85 (gpm) (tube rupture)	NA
De-boration by pool clean up Ion Exchange	7 – 10 PPM / change	NA	NA

6.0 INSTRUMENTATION

6.1 Loss of Offsite Power and Impact on the Spent Fuel Pool Level Instrumentation

Instrumentation is provided which monitors the temperature and the water level in both the refueling and spent fuel pools. Spent fuel pool alarms are annunciated locally and in the control room. Refueling pool computer alarms are annunciated in the control room only. Additional instrumentation, monitored locally, is provided to check inlet and outlet temperatures on the heat exchangers, and to determine the pressure of the cooling pump discharge. The power source for instrumentation is provided by normal 120 VAC (Panel E- NNN – D015). Automatic transfer to back up reliable power source is available. Additionally, in the event that the distribution panel E-NNN-D15 is lost due to shedding of MCC E-NHN-M19 in a loss of all power event, provisions are made to enable operators to load key instruments on emergency diesels (Ref. 55 through 58). Procedures 40AO-9ZZ15 and 40AO-9ZZ12 provide instruction for the implementation of the design and to ensure that loss of annunciators, and degraded electrical power conditions are addressed (Ref. 59 and 60).

[1] 500 gpm is the maximum flow of a fire hose station. 45 gpm, is estimated flow for sprinkler system.

6.2 Loss Offsite Power and Impact on Borated Make–up System Instrumentation

RWT Level

The RWT level is measured and monitored in the Control Room by six separate level instrumentation loops (Ref. 12). These instrument loops are designed to Quality Class-Q, Seismic Category I requirements. Four of the level instrument channels are designated as CHA-LI-203A, CHB-LI-203B, CHC-LI-203C & CHD-LI-203D (Ref. 61 & 62). The Class 1E Instrumentation Distribution Panels, PNA-D25, PNB-D26, PNC-D27 & PND-D28, respectively power these level monitoring loops (Ref. 63 & 64). These indicators are located on Control Board B02D in the Control Room. The two additional level instruments are designated as CHA-LI-200 and CHB-LI-201. These two level monitoring loops are powered by the Class 1E instrumentation distribution panels PNA-D25 & PNB-D26 respectively. These indicators are located on Control Board B03A in the Control Room.

BAMPs

The Seismic Category I piping associated with the BAMP makeup flow path to the SFP does not include instrumentation for flow rate measurements. However, the pump discharge pressure is available locally at the BAMPs on pressure gauges CHN-PI-206-1 & CHN-PI-207-1 (Ref. 12). In addition, Control Room indication of the BAMP discharge pressure is provided on Control Board B03A by indicators CHN-PI-206 and CHN-PI-207 (Ref. 66). These pressure indicators are designed to Quality Classification QAG and Seismic Category 9 requirements. The Control Room indicators are powered by 120V AC Non-Class 1E Ungrounded Instrument & Control Panel E-NNN-D12 (Ref. 68). Backup Class 1E power is provided to this Non-Class 1E instrument panel through a Static Transfer Switch. The Static Transfer Switch is designed to allow the transfer of the load to the Class 1E source on a loss of the Non-Class 1E supply. The Class 1E backup power is provided by Motor Control Center E-PHB-M32.

BAMP Boration of the SFP through the BABT

This boration flow path utilizes portions of the Non-Seismic Category I piping, valves and components associated with the BABT alignment. The BAMP discharge pressure is available as noted above. The flow rate from the BAMPs is measured and manually adjusted from the Control Room utilizing Boric Acid Makeup Controller CHN-FIC-210Y (Ref. 12). This instrument is designed to Quality Classification QAG and Seismic Category 9 requirements. This control loop is powered by 120V AC Non-Class 1E Ungrounded Instrument & Control Panel E-NNN-D11 (Ref. 69). Backup Class 1E power is provided to this Non-Class 1E instrument panel through a Static Transfer Switch. The Static Transfer Switch is designed to allow the transfer of the load to the Class 1E source on a loss of the Non-Class 1E supply. The Class 1E backup power is provided by Motor Control Center E-PHA-M31 (Ref. 70).

RWMPs

As noted previously, the RWMPs can be utilized to provide non-borated makeup water to

the SFP from the RWMT. The pump discharge pressure is available locally at the RWMPs on pressure gauges CHN-PI-208-1 & CHN-PI-209-1 (Ref. 71). These indicators are designed to Quality Classification NQR and Seismic Category 3 requirements. In addition, Control Room indication of the RWMP discharge pressure is provided on Control Board B03A by indicators CHN-PI-208 and CHN-PI-209 (Ref. 67). These pressure indicators are designed to Quality Classification QAG and Seismic Category 9 requirements. These indicators are powered by 120V AC Non-Class 1E Ungrounded Instrument & Control Panel E-NNN-D12 (Ref. 68). Backup Class 1E power is provided to this Non-Class 1E instrument panel through a Static Transfer Switch. The Static Transfer Switch is designed to allow the transfer of the load to the Class 1E source on a loss of the Non-Class 1E supply. The Class 1E backup power is provided by Motor Control Center E-PHB-M32.

This makeup flow path utilizes portions of the Non-Seismic Category I piping, valves and components associated with the BABT. The makeup flow rate from the RWMPs is measured and manually adjusted from the Control Room utilizing Reactor Makeup Water Flow Controller CHN-FIC-210X (Ref. 12). This controller is designed to Quality Classification NQR and Seismic Category 3 requirements. This control loop is powered by 120V AC Non-Class 1E Ungrounded Instrument & Control Panel E-NNN-D11 (Ref. 69). Backup Class 1E power is provided to this Non-Class 1E instrument panel through a Static Transfer Switch. The Static Transfer Switch is designed to allow the transfer of the load to the Class 1E source on a loss of the Non-Class 1E supply. The Class 1E backup power is provided by Motor Control Center E-PHA-M31 (Ref. 70).

RWMT Level

The RWMT is equipped with two separate level instruments. Local level indication is provided at the tank by CHN-LI-211. This level indicator is designed to Quality Classification NQR and Seismic Category 3 requirements. Control Room indication of the RWMT level is provided on Control Board B03A by level indicator CHN-LI-210 (Ref. 71). The transmitter in this loop is designed to Quality Classification NQR and Seismic Category 3 requirements while the Control Room indicator meets Quality Classification QAG requirements. This control loop is powered by 120V AC Non-Class 1E Ungrounded Instrument & Control Panel E-NNN-D11 (Ref. 72 & 73). Backup Class 1E power is provided to this Non-Class 1E instrument panel through a Static Transfer Switch. The Static Transfer Switch is designed to allow the transfer of the load to the Class 1E source on a loss of the Non-Class 1E supply. The Class 1E backup power is provided by Motor Control Center E-PHA-M31 (Ref. 70).

Safety Injection System

Both pump discharge pressure and flow rate indications are available in the Control Room for the Containment Spray pump makeup alignment to the SFP. Indicators SIN-PI-306 & PI-307 provide CS pump discharge pressure. Indicators SIN-PI-303X and PI-303Y provide additional pressure readings (Ref. 75). These indicators are referenced for the "A" and "B" SI Trains respectively. These instruments are designed to Quality Classification QAG and Seismic Category 9 requirements. These instrument loops are powered by the120V AC Non-Class 1E Ungrounded Instrument & Control Panels

E-NNN-D11 & D12 similar to the BAMP and RWMP instrumentation described previously (Ref. 68 & 70).

Flow indicators SIA-FI-338 & SIB-FI-348 provide Control Room flow indication for the CS pump "A" and "B" Trains respectively (Ref. 75). These instrument loops are designed to Quality Class Q and Seismic Category I requirements (Ref. 75). The Class 1E Instrumentation Distribution Panels, PNA-D25 & PNB-D26 respectively power these flow-monitoring loops (Ref. 62 & 74). These indicators are located on Control Boards B02E and B02D in the Control Room.

The pump discharge pressure is available in the Control Room for the LPSI pump makeup flow path to the SFP. The indicators SIN-PI-306 & PI-307, described above for the CS pumps, provide pressure indication for this alignment also. There is no flow indication either locally or in the Control Room for the LPSI pump makeup flow path to the SFP.

6.3 Loss of Offsite Power and Impact on Other Systems Instrumentation

Instrumentation on all non-quality systems such as nuclear cooling water, liquid radwaste, Domestic water, and Demineralized water could be lost during total loss of power. These systems instrumentation are not critical during mitigation of boron dilution event.

Therefore essential instrumentation on this system could be available. Instrumentation for Condensate Storage and Transfer System and Essential Cooling Water system are available during total loss of power and backup by a 1E power source.

7.0 SPENT FUEL POOL DILUTION EVALUATION

For the purposes of evaluating spent fuel pool dilution times and volumes, the total pool volume available for dilution is conservatively assumed to be 340,000 gallons (Ref. 76), for normal plant operation (operator error), fire event and pipe breaks. The volume of transfer canal and cask loading pit are 34,000 and 82,000 gallons respectively. During normal plant operation the cask-loading pit is filled with borated water and the transfer canal may be filled and, they are isolated from the main pool by pneumatic sealed gates. The normal configuration of the spent fuel pool is to have all of the gates in place. In this configuration, any dilution of the pool, the transfer canal or cask loading pit is assumed to affect only the body of water that non-borated water is introduced in.

During a seismic event, the total pool volume available for dilution is conservatively assumed to be 206,000 gallons (Ref. 77). During this event, due to the failure of non-quality pool gates and pool clean up system, all bodies of water would reach equilibrium at elevation 133ft. The initial conditions for the event are assumed to be as follows; spent fuel pool water level is at 137 ft, the Cask loading pit is only filled to the 133 ft elevation, and the transfer canal is empty prior to the event. In this configuration, any dilution in the spent fuel pool, the transfer canal or cask loading pit is assumed to affect all bodies of water that non-borated water is introduced.

The boron concentration currently maintained in the spent fuel pool is 4,000 – 4,400 ppm for plant Modes 1 through 6. In addition, the boron concentration is maintained at greater than

2,150 ppm when irradiated fuel is stored in the spent fuel pool. Based on newly proposed Technical Specification, LCO 3.7.15, the minimum allowable soluble boron concentration required to maintain the spent fuel boron concentration at $K_{eff} < 0.95$, including uncertainties and burnup, with a 95% probability at a confidence level (95/95) is 900 ppm.

For the purposes of the evaluating dilution times and volumes, the initial spent fuel pool boron concentration is assumed to be at the current Technical Specification Limit of 2,150 ppm. The evaluations are based on the spent fuel pool boron concentration being diluted from 2,150 ppm to 900 ppm. To dilute the spent fuel pool volume of 320,000 gallons during normal operation, from 2,150 ppm to 900 ppm, it would conservatively require 7.63E+5 gallons of non-borated water. During seismic event, the volume of water in the pool would be approximately 206,000 gallons. To dilute the spent fuel pool from 2,150 ppm to 900 ppm, it would conservatively require 4.91E+5 gallons of non-borated water.

This analysis assumes thorough mixing of all the non-borated water added to the spent fuel pool. It is unlikely, with cooling flow and convection from the spent fuel decay heat, that thorough mixing would not occur. However, if mixing were not adequate, it would be conceivable that a localized pocket of non-borated water could form somewhere in the spent fuel pool. This possibility is addressed by the criticality calculation, which shows that the spent fuel K_{eff} will be less than 1.0 on a 95/95 basis with the spent fuel pool filled with non-borated water. Thus, even if a pocket of non-borated water formed in the spent fuel pool, K_{eff} would not be expected to exceed 1.0 anywhere in the pool.

The time to dilute ($T_{dilution}$) depends on the initial volume of the pool and the postulated rate of dilution. The dilution volumes and times for the dilution scenario discussed in Sections 3.2 and 3.3 are calculated based on the following equation:

$$T_{dilution} = (V/Q)^* \text{Ln}\,(C_o/C_{end}) \qquad \text{(Equation 1)}$$

Where:

C_o = the boron concentration of the pool volume at the beginning of the event (2,150 ppm)

C_{end} = the boron endpoint concentration (900 ppm)

Q = dilution rate (gallons of water/minute)

V = volume (gallons) of spent fuel pool.

7.1 Spent Fuel Dilution During Seismic Event (Power Operation / Refueling)

7.1.1 Description of Event

In hypothetical event of SSE with loss of offsite power, the spent fuel pool water level could drop due to the failure of non- seismic pool clean up system and gates. The minimum elevation reached during this event is adequate to maintain pool cooling and provide more than 10-ft. of shielding above the fuel assemblies. Post seismic, the spent fuel pool, cask loading pit and transfer cannel reach equilibrium. The design leakage due to evaporation is less than

2 gpm. Liner and boundary valve leakage is limited to less than 5 gpm during plant modes 1 through 5. During modes 5 and 6 the design leakage from pool boundary is limited to less than 30 gpm when valve PCN–V118 is in close position and refueling pool level is less than 133 ft. The normal source of make up during this mode of operation is the gravity feed from the refueling tank, as discussed in section 4.1 of this report.

7.1.2 Calculation of Boron Dilution Times and Volumes

Based on design bases of PVNGS (Ref. 78): Non-safety systems are assumed not available to mitigate accident conditions but are not considered to operate in a manner which increases the severity of the accident. Therefore, it is assumed all non–seismic systems such as domestic, demineralized water, fire protection, Condensate transfer, liquid radwaste will fail. These systems would not contribute to the event since the motive force of the supply sources (i.e. supply pumps and tanks [2]) would fail as result of seismic event. As described in section 5 of this report, the lines providing water to the utility / decontamination and fire hose stations terminate at elevation 144 ft. There would be minimum amount of water spilled due to failure of seismic IX piping. The volume of water spilled during event from non-borated source is estimated to less than 15 gallons (100 ft. of 1 ½ inch piping).

The makeup source for this event is the Refueling Water Storage Tank. As stated in section 7.1.1, the gravity feed would be able to makeup with system and boundary leakage in all modes of operation. It should be noted that during modes 5-6 operation when refueling pool level is less than 133 ft., there will be more than 400,000 gal loss of borated water available to compensate for the design leakage.

7.1.3 Evaluation of Boron Dilution Event

It can be concluded for the above discussion that seismic event would not result in a boron dilution event. The addition of 15 gallons non-borated water in 206,000 gallons of borated inventory @ minimum 2150-ppm is insignificant. Additional safety margin has been added to the design by administrative requirements of the procedure 4XA1-XRK7C. The control room operators would identify and isolate sources of leakage in a timely, once manner a valid seismic alarm is received.

7.2 Moderate Energy Line Break

7.2.1 Description of Event

The plant design for protection against piping failures in the fuel building is reviewed to assure that such failures would not cause de-boration of the spent

[2] Note: All supply lines are isolated by manual root valves in the fuel building. The CST and liquid radwaste system are design to seismic category II, and would retain their pressure boundary integrity up to OBE event.

fuel pool and to assure that the spent fuel pool boron concentration will remain above minimum required. The review includes moderate energy fluid system piping located within the fuel building. It is assumed for this analysis that a hypothetical random pipe break would occur within the fuel building.

The pipe break is the initiating event, and it does not coincide with any other design bases events. As discussed in section 5 of this report all lines within the fuel building can be classified as moderate energy lines. Per ANS/ANSI 58.2–1980, through–wall crack opening shall be assumed as a circular orifice of cross-section flow area equal to that of rectangle with length of ½ the pipe inside diameter and with of ½ the pipe thickness. Table 4 shows a summary of calculated leakage for different systems. The hypothetical pipe break does not result in loss of equipment power or flooding of SSC. Therefore, no loss of offsite power is required for this event.

7.2.2 Calculation of Boron Dilution Times and Volumes

As described in section 5 and shown on Table 3, the most limiting pipe leakage is due to a failure in the fire protection piping. The fire protection hose station is located at elevation 140 ft. (pool operating deck). It is conservatively assumed that all 85-gpm discharged from the crack would be added to the pool (due to the geometry of the fuel building, a large portion of for the leakage will spill on to lower floors). Since the initiating condition would happen during normal operation, the total volume of the spent fuel pool is available for dilution.

Therefore based on equation one:

- C_o = the boron concentration of the pool volume at the beginning of the event = 2150 ppm (assuming smallest pool concentration)
- C_{end} = the boron endpoint concentration = 900 ppm
- Q = dilution rate (gallons of water/minute) = 85 gpm
- V = volume (gallons) of spent fuel pool = 320,000 gals

Using equation one, the time to dilution is calculated to be approximately 55 hr. This would provide adequate time for operators to respond to the event. The pool Hi-Hi level alarm is currently set at 138' 4". Assuming the initiation event accrues at the time the pool has minimum possible Technical Specification elevation, i.e. 137' 4", the time for the operator to get an indication in control room is approximate 1.1 hr and the pool concentration at time of alarm would be greater than 2,050 ppm. Isolation of the header is easily achievable since the isolation valve is located outside the fuel building (PIV-014).

7.2.3 Evaluation of Boron Dilution Event

It can be concluded, from the above discussion, that a pipe break event would not result in a boron dilution event, which could reduce the margin of safety. The addition of 85 gpm non-borated water to the spent fuel pool is at a flow rate that can be identified and isolated by control room operators in a reasonable time period.

7.3 Normal Operation, Operator Error and Fire

The scenarios evaluated in this section can be categorized into three categories;

1. Normal operational occurrences which include normal replacement of ion exchanges in the pool cooling system, normal evaporation makeup, and normal inter system leakages (ruptures),

2. Operator error which include misuse of decontamination / utility station and

3. Fire in the fuel building

   a. Detail discussions of each system and normal parameters associated are providing in section 5. During normal operation the pool clean up system ion exchanges need to be replenished. This process would result in de-boration of the fuel pool. Since the normal boron concentration in the spent fuel pool is usually greater then 4,000 ppm. The effect of de-boration due to introduction of 850 gallons of demineralized water or de-boration due to ion-exchanger (7-10 ppm) would be insignificant.

      The inter-system leakage from nuclear cooling / essential cooling system into the pool cooling system adds an insignificant amount of non-borated water into the pool. At rate of 1 gpm, the leakage would be almost undetectable and could be masked by evaporation from the spent fuel pool. However, control room operators would observe excessive demand on makeup water required for the nuclear cooling / essential cooling water system. In event of inter system tube rupture, the flow initially would be approximately 85 gpm; however, closed loop-cooling system would lose pressure rapidly and the control room would have an indication due to Low-Low level alarms produced by surge tank levels. Such event is bounded by the evaluation performed for pipe break (see section 7.2).

   b. It is possible to assume that a single operator using the decon-station / utility station would by mistake discharge a portion of hose flow into the spent fuel pool during maintenance activities or decontamination of casks or adjacent areas around the pool. The flow form these stations are limited to approximately 50 gpm. At this flow rate, the discharge from the hose is less than in the case evaluated for a pipe break, therefore, it would be bounded by the evaluation in section 7.2. However, it is highly improbable that the flow would be continuous for long duration of time. Therefore, the consequences of such an event is much less than a moderate

energy line break event.

c. The probability of fire at the fuel building at elevation 140 ft. is very small due to the lack of combustible loading. In case a fire at this elevation, the fire team has to utilize the fire hose station HS 100, located at the southeast corner of the operating floor. The maximum flow for this hose station is 500 gpm and assuming that the fire can be terminated within an hour, the total water discharge to elevation 140-ft. would be 30,000 gal. If all flow is discharge directly into the pool, the pool concentration would be reduced from 2,150 to 1,900 ppm (note: normal concentration for spent fuel pool is greater than 4,000 ppm). Because of the limited flow into the spent fuel pool enclosure, and because of the awareness of control room and fire crew, the discharge to the pool would be terminated long before the spent fuel pool boron concentration could be reduced to 900 ppm,

7.4 Conclusions

It is concluded that an unplanned or inadvertent event, which would result in the dilution of the spent fuel pool boron concentration from 2,150 ppm to 900 ppm is not a credible event. This conclusion is based on the following:

1. In order to dilute the spent fuel pool ($K_{eff} > 0.95$), a substantial amount of water (nearly 490,000 gallons) is needed. Most sources of water at PVNGS site would be exhausted and it would take continued manual actions on the part of plant personnel to assure that enough water would be available to support such a dilution.

2. Since such a large water volume turnover is required, a spent fuel pool dilution event would be readily detected by plant personnel via alarms, sump flooding in the fuel and / or auxiliary building or by normal operator rounds through the spent fuel pool area.

3. Evaluations indicate that based on the design flow rates of non-borated water normally available to the spent fuel pool, sufficient time is available to detect and respond to such an event.

It should be noted that this boron dilution evaluation was conducted by evaluating the time and water volumes required diluting the spent fuel pool from 2,150 ppm to 900 ppm. The 900-ppm end point was utilized to ensure that K_{eff} for the spent fuel would remain less than or equal to 0.95. However, the PVNGS technical requirement manual requires the spent fuel pool to be maintained at concentration of 4,000 to 4,400 ppm boron. This requirement would provide additional design margin, which is not credited for in this study. In conclusion, the design and administrative procedures in place at PVNGS provide solid design bases for crediting soluble boron in the spent fuel pool and the plant design provides ample margin against an inadvertent dilution event.

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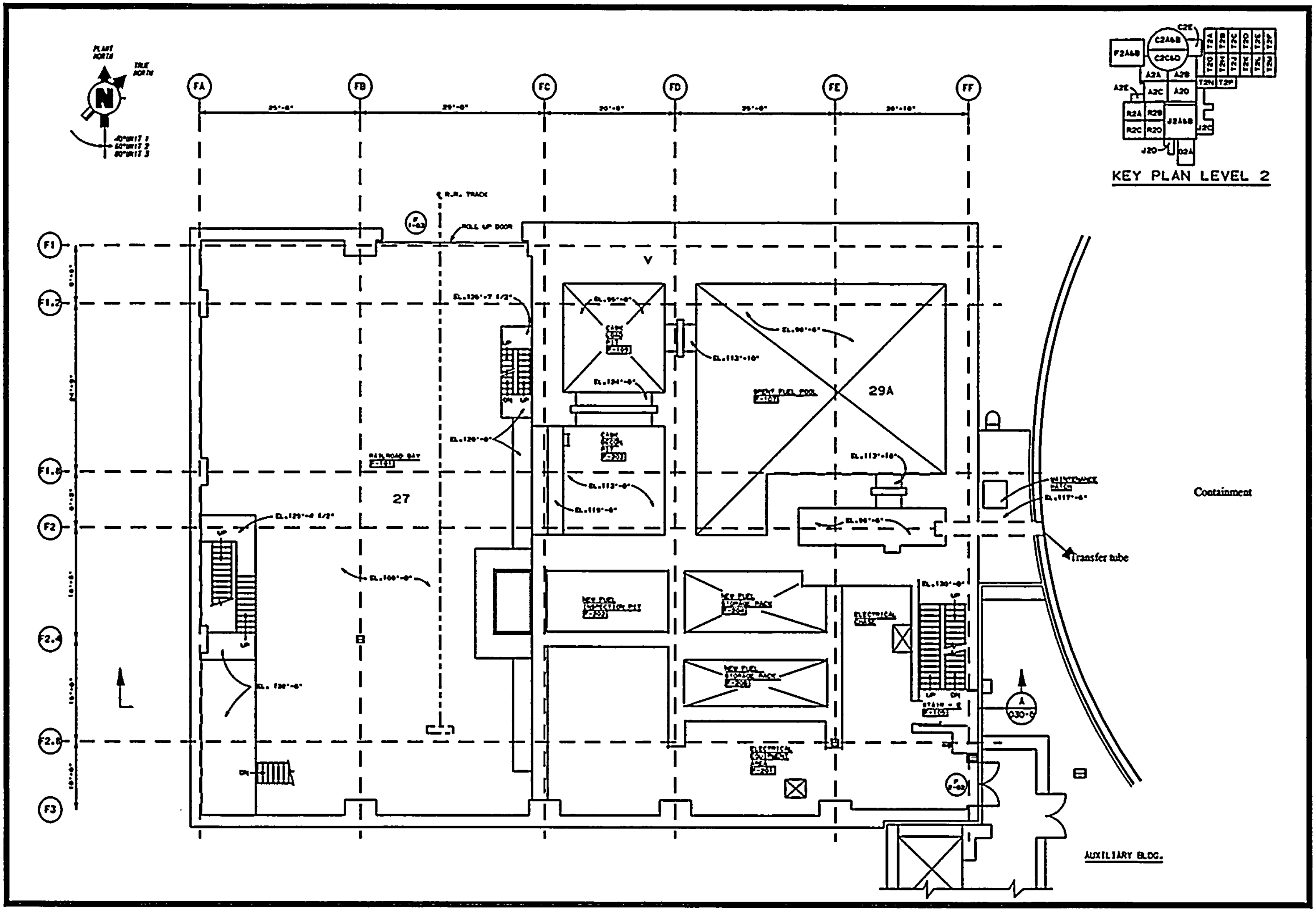


Figure 1. Fuel Building arrangement at plan elev. 120 ft.

Figure 2. Spent fuel pool and refueling pool physical arrangement

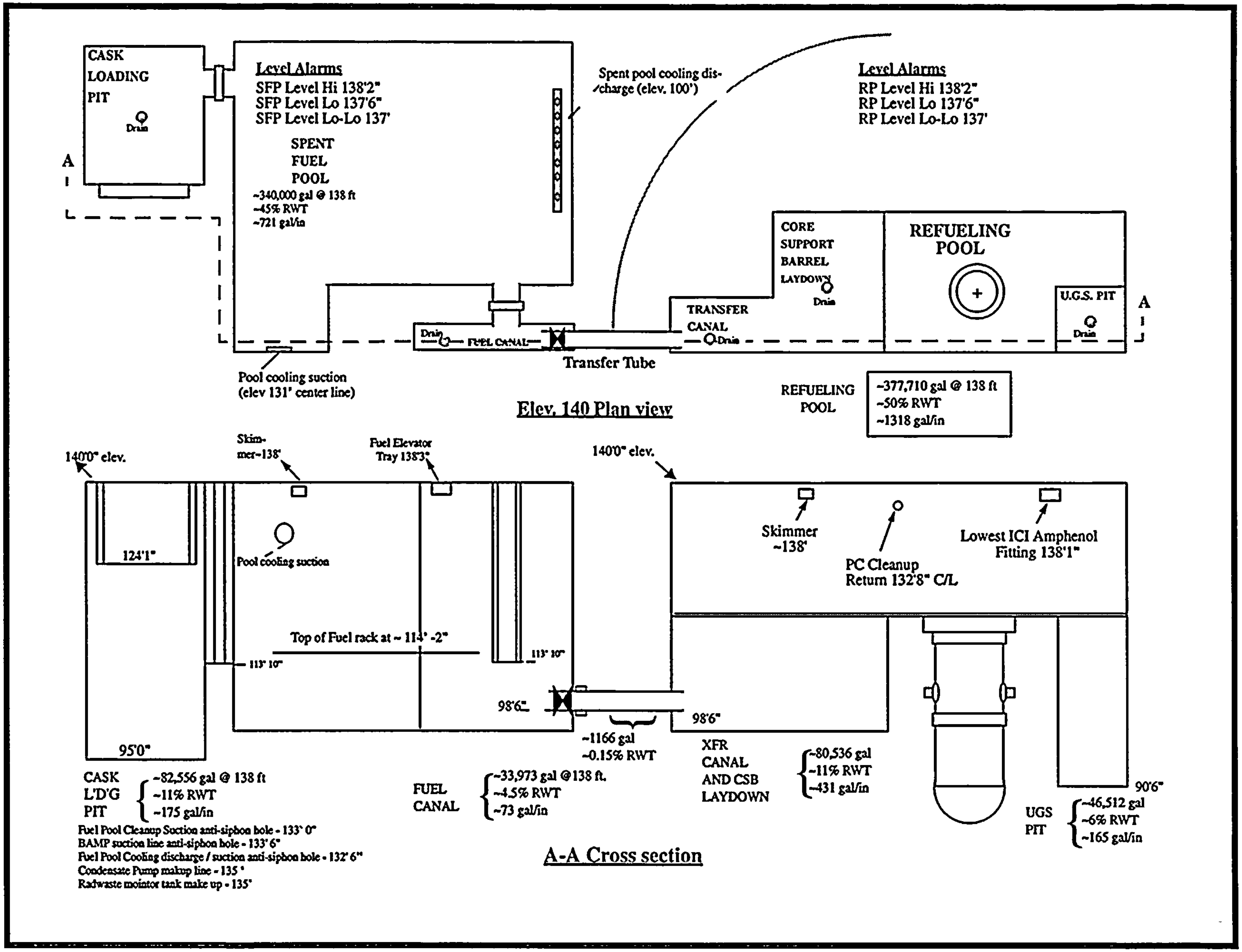
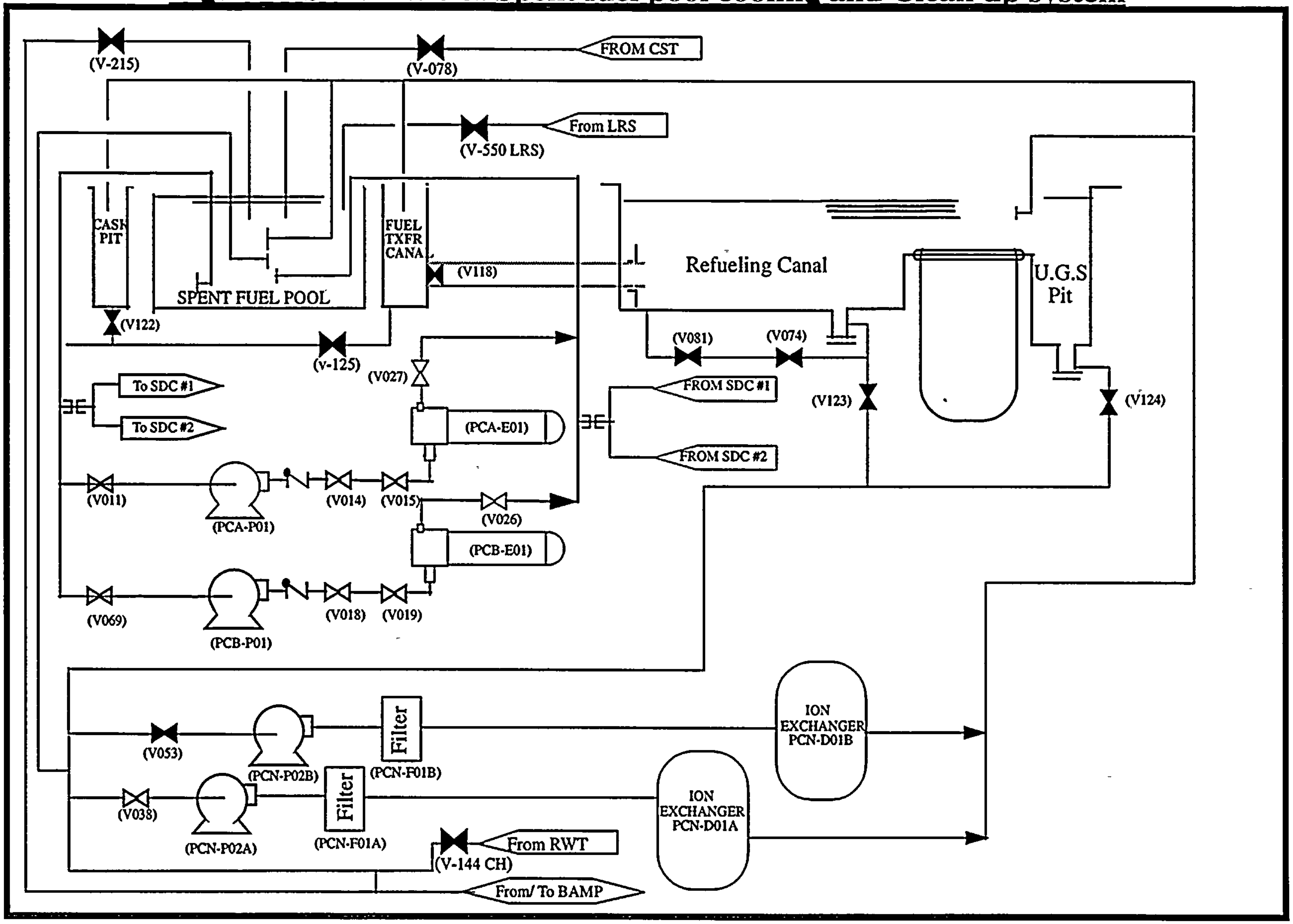


Figure 3. Schematic of Spent fuel pool cooling and Clean up system



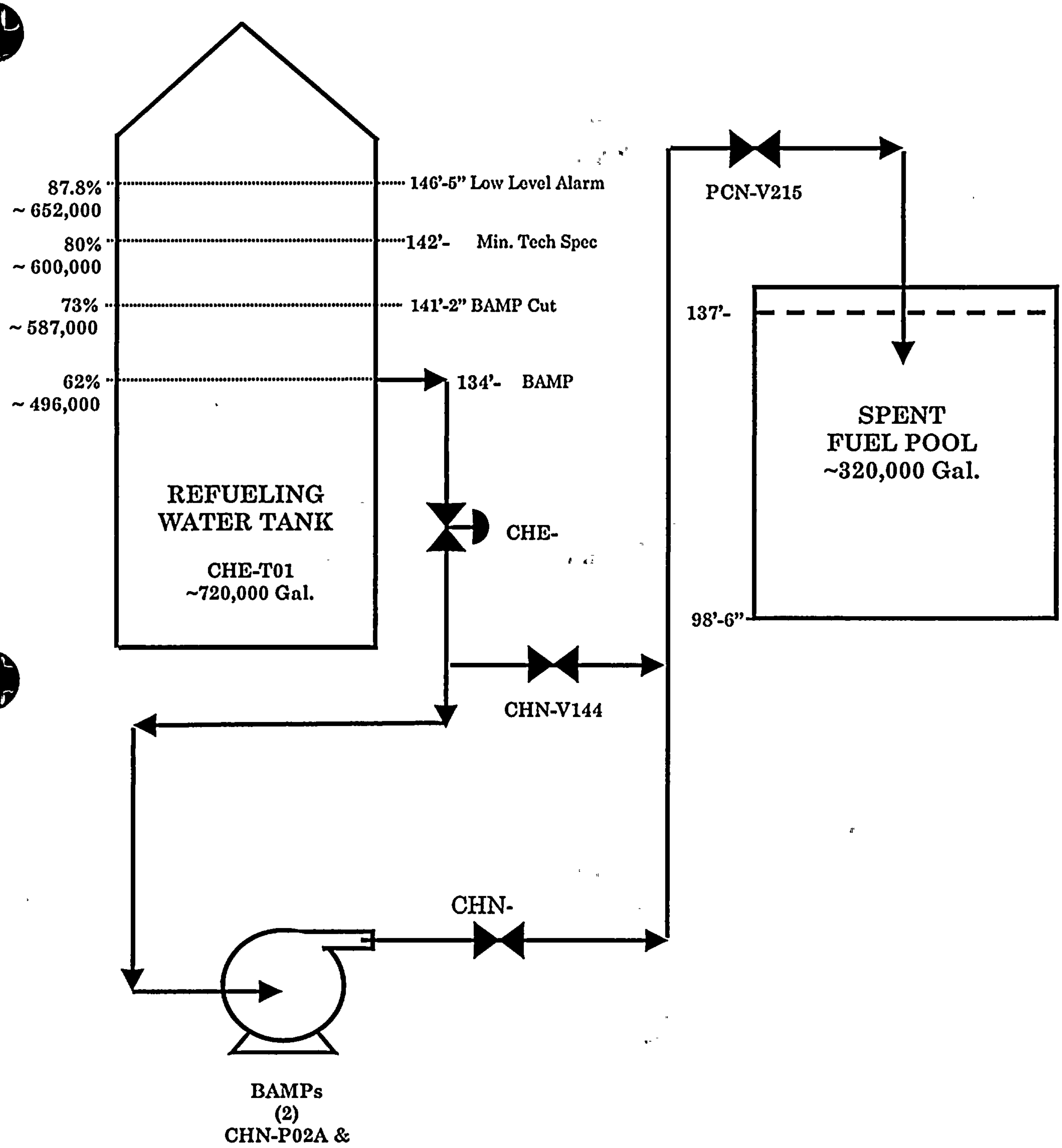


Figure 4. Spent Fuel Pool Makeup Flow Paths, BAMPs & Gravity Feed Flow

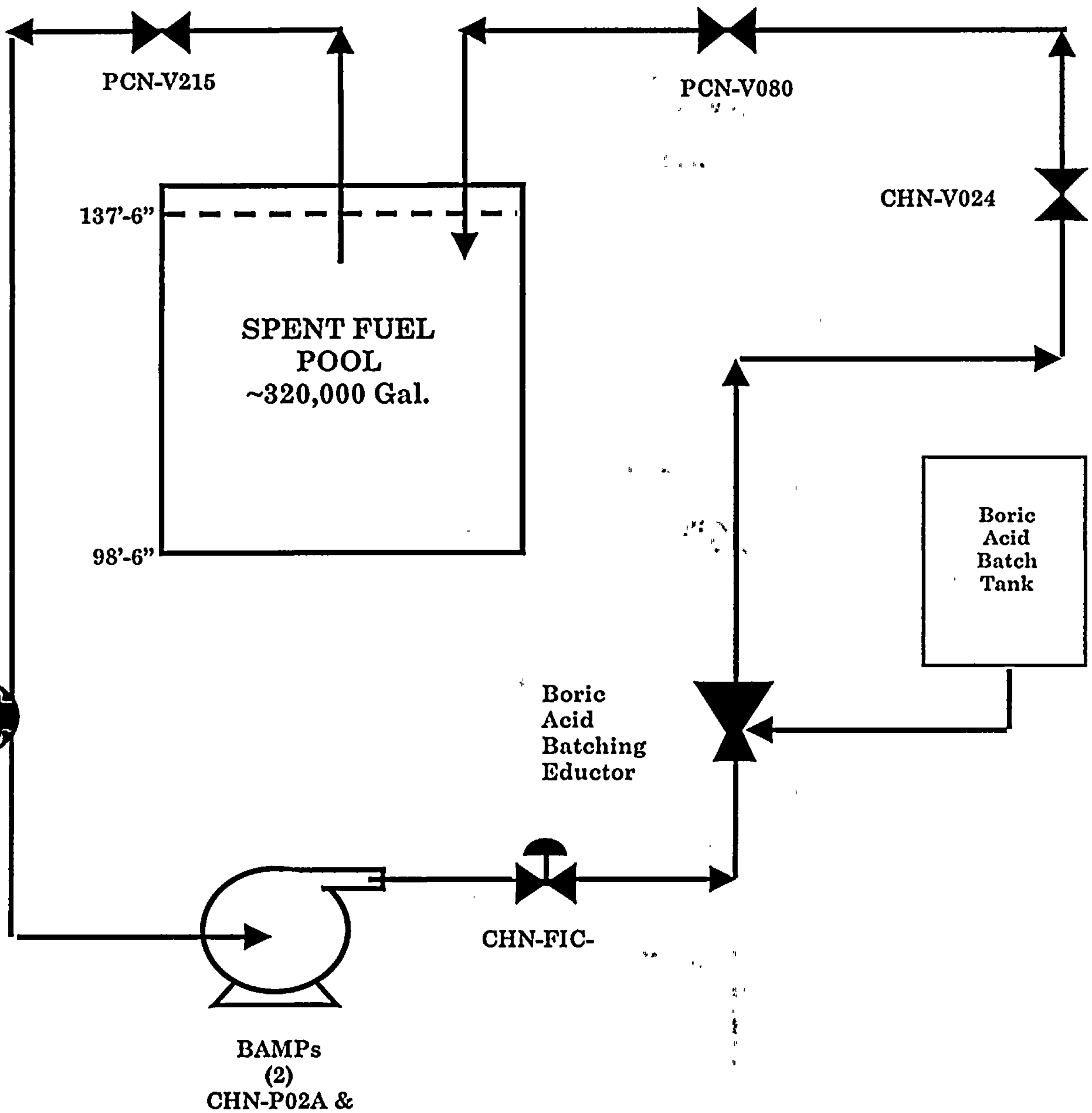


Figure 5. Spent Fuel Pool Boration
Flow Path, BAMPs via BABT

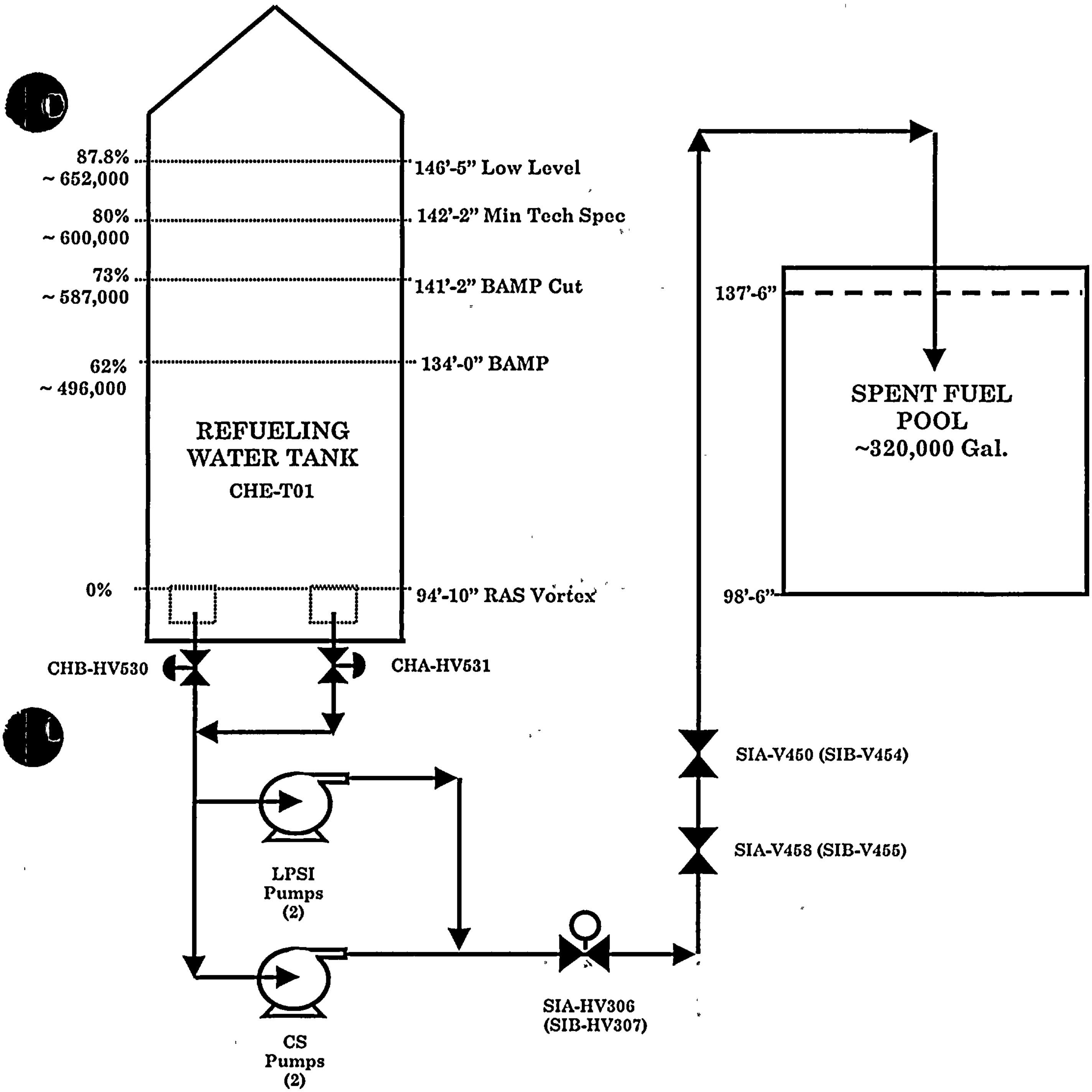


Figure 6. Spent Fuel Pool Makeup

ENCLOSURE 5

Additional Information Used in the Palo Verde Spent Fuel Pool Criticality Analysis

Enclosure 5 - Additional Information Used in the Palo Verde Spent Fuel Pool Criticality Analysis

Question: For which specific actinides and fission products is burnup credit given?

Response: The fuel assembly depletion calculations were performed with the two-dimensional multi-group transport code DIT, using an 89-group ENDF/B-VI based cross section library. The DIT cross section library includes 21 actinide and 116 fission product nuclides, which contribute to the reactivity loss with burnup. The depletion chains used to generate nuclide concentrations for the spent fuel pool criticality analysis are the same as those used in nuclear design, which have been benchmarked through core follow analysis. The lists of actinides and fission products are given in the following tables. Samarium and Promethium isotopes of mass number 149 are explicitly modeled with a yield due to direct fission (denoted by the "D" suffix in the table) and a yield due to production from other isotopic decay.

Actinides and Fission Products Included in DIT Depletions with ENDF/B-VI Cross Sections

Actinides

92-U -235	94-PU-242
92-U -236	95-AM-241
92-U -237	95-AM-242
92-U -238	95-AM-242M
93-NP-237	95-AM-243
93-NP-238	96-CM-242
93-NP-239	96-CM-243
94-PU-238	96-CM-244
94-PU-239	96-CM-245
94-PU-240	96-CM-246
94-PU-241	

Fission Products

35-BR- 81	44-RU-105	53-I -129	59-PR-143	63-EU-151
36-KR- 83	44-RU-106	53-I -131	60-ND-142	63-EU-153
36-KR- 85	45-RH-103	53-I -135	60-ND-143	63-EU-154
38-SR- 89	45-RH-105	54-XE-131	60-ND-144	63-EU-155
38-SR- 90	46-PD-104	54-XE-132	60-ND-145	63-EU-156
39-Y - 89	46-PD-105	54-XE-133	60-ND-146	63-EU-157
39-Y - 91	46-PD-106	54-XE-134	60-ND-147	64-GD-154
40-ZR- 91	46-PD-107	54-XE-135	60-ND-148	64-GD-155
40-ZR- 93	46-PD-108	54-XE-136	60-ND-150	64-GD-156
40-ZR- 95	47-AG-109	55-CS-133	61-PM-147	64-GD-157
40-ZR- 96	47-AG-110M	55-CS-134	61-PM-148	64-GD-158
41-NB- 95	47-AG-111	55-CS-135	61-PM-148M	65-TB-159
42-MO-95	48-CD-110	55-CS-136	61-PM-149D	65-TB-160
42-MO- 96	48-CD-111	55-CS-137	61-PM-149	65-TB-161
42-MO- 97	48-CD-113	56-BA-134	61-PM-151	66-DY-160
42-MO- 98	49-IN-115	56-BA-137	62-SM-147	66-DY-161
42-MO- 99	51-SB-121	56-BA-140	62-SM-148	66-DY-162
42-MO-100	51-SB-123	57-LA-139	62-SM-149D	66-DY-163
43-TC- 99	51-SB-125	57-LA-140	62-SM-149	66-DY-164
44-RU-100	51-SB-127	58-CE-141	62-SM-150	67-HO-165
44-RU-101	52-TE-127M	58-CE-142	62-SM-151	
44-RU-102	52-TE-129M	58-CE-143	62-SM-152	
44-RU-103	52-TE-132	58-CE-144	62-SM-153	
44-RU-104	53-I –127	59-PR-141	62-SM-154	

Question What additional uncertainties are assumed when credits for actinide and fission product decay are calculated?

Response: One additional uncertainty was considered for actinide and fission product decay. The burnup credit resulting from the decay of Pu-241 and build-up of Am-241 was reduced by 25 percent to conservatively account for reactivity "end-effects". The reactivity "end-effect" is the reactivity associated with the decay of extremely limiting Pu-241 and Am-241 axial distributions not normally encountered in typical PWR fuel management schemes. The decay calculations were performed with the DIT code, using the set of actinide and fission product nuclides described above. The short lived nuclides such as I-131, Xe-131, Pm-149 and NP-239 were decayed for all pool criticality calculations. For long cooling periods, the decay of Pu-241 into Am-241 is by far the most important transition.

Because all Palo Verde Units are operated with an unrodded core, exposures of discharged assemblies are characterized by nearly symmetric axial burnup distributions that have a slight top peaked bias. As a result, there are minor differences between axially uniform isotopic distributions and the actual assembly nuclide distributions. A slightly top peaked Pu-241 distribution will decay into a slightly top peaked Am-241 distribution. This non-uniformity in isotopic distribution results in a small non-conservative reactivity effect relative to the uniform axial distributions that were assumed in decay calculations.

Other conservatisms embedded in the model more than compensate for any postulated axial isotopic distribution effect. For instance, the assumption used in the depletion of a constant 1000 ppm soluble boron and 1200 °F fuel temperature results in a hardened spectrum which leads to an overprediction of the fissile content and to an overprediction of the reactivity in the spent fuel pool configuration. Typical reactivity effects after a 15 year cooling period are as follows:

Enrichment (w/o)	Exposure (MWD/T)	Conservatism due to nominal PPM & T_{fuel} % delta k	Non-conservatism due to axial Pu-241 redistribution at 15 yrs % delta k	Net Conservatism % delta k
3.0	22,400	0.6	0.4	0.2
4.5	40,000	0.6	0.5	0.1

Question How do the characteristics of the critical experiments used for benchmarking compare with those of the Palo Verde fuel and spent fuel storage racks?

Response The fuel assembly design used at all the Palo Verde Units is the 16 x 16 ABB-CE fuel assembly design for 150.0-inch active height cores. The fuel rods consist of sintered UO2 pellet columns encapsulated in zircaloy; the assumed UO2 maximum feed enrichment is 4.8 weight percent U-235. The fuel assembly storage rack modules are the monolithic design and employ 0.12 inch thick SS-304 box walls. A 0.175-inch thick SS-304 L-insert is employed in selective storage cells.

Selected lattices from two experimental programs were employed in the benchmarking analyses: (1) the Babcock & Wilcox program on Critical Experiments Supporting Close Proximity Storage of Power Reactor Fuel and (2) the Pacific Northwest Laboratory program on Criticality Experiments with Subcritical Clusters of 2.35 and 4.31 weight percent U-Enriched UO2 Rods in Water with Steel Reflecting Walls. More up-to-date information on material compositions for the latter experimental program

was obtained from documentation in the International Handbook of Evaluated Criticality Safety Benchmark Experiments prepared by the Nuclear Energy Agency and the Organization for Economic Cooperation and Development.

The experiments from these two programs were selected for the benchmarking of the SCALE methodology employed in analyzing the Palo Verde storage rack based on the following key parameters which span the variables of interest.

1) The fuel rods employed in the experiments contain sintered UO2 and employ low absorption clad tubes;
2) The enrichments of the fuel rods employed in the critical experiments are 2.549 weight percent for the B&W fuel and 4.306 weight percent for the PNL fuel, thus providing a close approximation to the range of interest for the storage rack analyses;
3) The B&W experiments employ a 3 x 3 array of 14 x 14 fuel rod clusters and the PNL experiments employ a 2 x 2 array of rod clusters with typical rod cluster sizes of 9 x 12 or 11 x 14 rods;
4) Eight of the B&W experiments employ 0.188-inch thick SS-304 isolation sheets in the water channels separating the rod clusters in the 3 x 3 array and two of the PNL lattices employ either 0.191-inch or 0.119-inch thick SS-304 isolation sheets in the cross shaped channel separating the rod clusters in the 2 x 2 array;
5) The PNL experiments extrapolate to critical on number of fuel rods with non-borated moderator whereas the B&W experiments employ fixed numbers of fuel rods in each lattice and go critical on water height for a given soluble boron concentration.

Therefore, the geometry and composition of the critical experiments provides a good match to the Palo Verde spent fuel pools.

Question The usual allowance for possible uncertainties in the fuel depletion analysis is 0.01 ΔK at 30,000 MWD/MTU applied linearly (5% of the reactivity decrement to the burnup of interest). Justify your use of 0.005 ΔK.

Response: The reactivity of the spent fuel pool as a function of burnup depends on the input nuclide concentrations as a function of burnup and on the microscopic cross sections assigned to these nuclides. The evolution of the nuclide concentrations was calculated by the two-dimensional multi-group transport code DIT, with an 89 group cross section library based on ENDF/B-VI. Benchmarking of the ROCS-DIT design codes using ENDF/B-VI cross sections for several operating cycles has indicated that

no reactivity bias is required in order to accurately predict the reactivity depletion rate (boron letdown curve) and the cycle length. This lends confidence to the nuclide concentrations generated for spent fuel pool criticality analyses.

Two conservatisms were also included in the nuclide concentration determination for Palo Verde. The first conservatism lies in the soluble boron concentration used in the DIT depletions. The assembly DIT depletions were performed at a soluble boron concentration of 1000 ppm. The high boron concentration results in a hardened spectrum, magnifies the conversion ratio and results in an increased fissile content at discharge. Therefore, the assembly reactivity will be conservatively overestimated when placed in the spent fuel pool.

The second conservatism lies in the fuel temperature used in the DIT depletion. A constant 1200 °F was assumed throughout the depletion, rather than the more realistic relationship where fuel temperature decreases with increasing burnup. The high fuel temperature increases the U-238 resonance absorption, increases the conversion ratio and magnifies the fissile content. From the above discussion, it can be concluded that the fissile content will contribute to a slight overprediction of the reactivity. A portion of these conservatisms have been used to address a reactivity effect associated with Pu-241 decay.

An additional component of the spent fuel pool reactivity lies in the cross sections used in KENO. KENO uses a 44 group cross section library based on ENDF/B-V. A comparison of DIT and KENO reactivities for the same set of nuclide concentrations was performed for a pin cell and a whole assembly. The most straightforward comparison between the DIT and KENO cross section libraries is through a pin cell reactivity calculation. This comparison was done for both a fresh fuel pin enriched to 4.5 weight percent, and a fuel pin having an initial enrichment of 4.5 weight percent depleted to 40,000 MWD/T

Using the same set of nuclide concentrations, the KENO pin cell shows a reactive gain over the DIT pin cell of 276 pcm by 40,000 MWD/T. The KENO whole assembly calculations assumed that the pin burnup distribution was flat, i.e. there was no provision to account for the water hole peaking factors. This assumption is also conservative, because the fuel rods adjacent to the water holes exhibit a higher burnup and a lower k-infinity. These pins have a higher importance in the assembly because of the softer spectrum and higher flux induced by the large water holes. Therefore, an assembly modeled with a uniform burnup will be more reactive than an assembly having a distributed burnup at 40,000 MWD/T.

It can be seen that KENO becomes increasingly more reactive than DIT

as one transitions from a fresh pin cell to a depleted pin cell and to a depleted assembly. Having established the good reactivity performance of DIT as a function of burnup, one can conclude that KENO is conservative in its reactivity estimates for depleted fuel, and that the reactivity bias of 0.005 delta k per 30,000 MWD/T is sufficient.