3.2 Characteristics of SFP Design and Operations for a Decommissioning Plant

Based on information gathered from the site visits and interactions with NEI and other stakeholders, the staff modeled the spent fuel pool cooling and cleaning (SFPC) system (see Figure 3.1).

- 2 redundent cooling pumps
- filtration subsystem
- ultimate heat sink is air
- manually operated makeup system (with a limited volumetric flow rate) supplements the small losses due to evaporation
- Back up makeup can use the firewater system, if needed. Two firewater pumps, one motor-driven (electric) and one diesel-driven, provide firewater in the SFP area. There is a firewater hose station in the SFP area. The firewater pumps are in a separate structure.

Based upon information obtained during the site visits and discussions with decommissioning plant personnel during those visits, the staff also made the following assumptions that are believed to be representative of a typical decommissioning facility:

- The SFP cooling design, including instrumentation, is at least as capable as that assumed in the risk assessment. Licensees have at least one motor-driven and one diesel-driven fire pump capable of delivering inventory to the SFP (SDA #1, Table 4.2-2).

- The makeup capacity (with respect to volumetric flow) is assumed to be as follows:

Makeup pump:	20 – 30 gpm
Firewater pump:	100 – 200 gpm
Fire engine:	100 – 250 gpm (100 gpm, for hose: 1½-in., 250 gpm for 2 1/2-in. hose)

- For the larger loss-of-coolant-inventory accidents, water addition through the makeup pumps does not successfully mitigate the loss of the inventory event unless the location of inventory loss is isolated.

- The SFP fuel handlers perform walkdowns of the SFP area once per shift (8- to 12-hour shifts). A different crew member works the next shift. The SFP water is clear and the pool level is observable via a measuring stick in the pool to alert fuel handlers to level changes.

- Plants do not have drain paths in their SFPs that could lower the pool level (by draining, suction, or pumping) more that 15 feet below the normal pool operating level, and licensees must initiate recovery using offsite sources.

Based upon the results of the June 1999 preliminary risk analysis and the associated sensitivity cases, it became clear that many of the risk sequences were quite sensitive to the performance of the SFP operating staff in identifying and responding to off-normal conditions. This is because the remaining systems of the SFP are relatively simple, with manual rather than automatic initiation of backups or realignments. Therefore, in scenarios such as loss of cooling or inventory loss, the fuel handler's responses to diagnose the failures and bring any available

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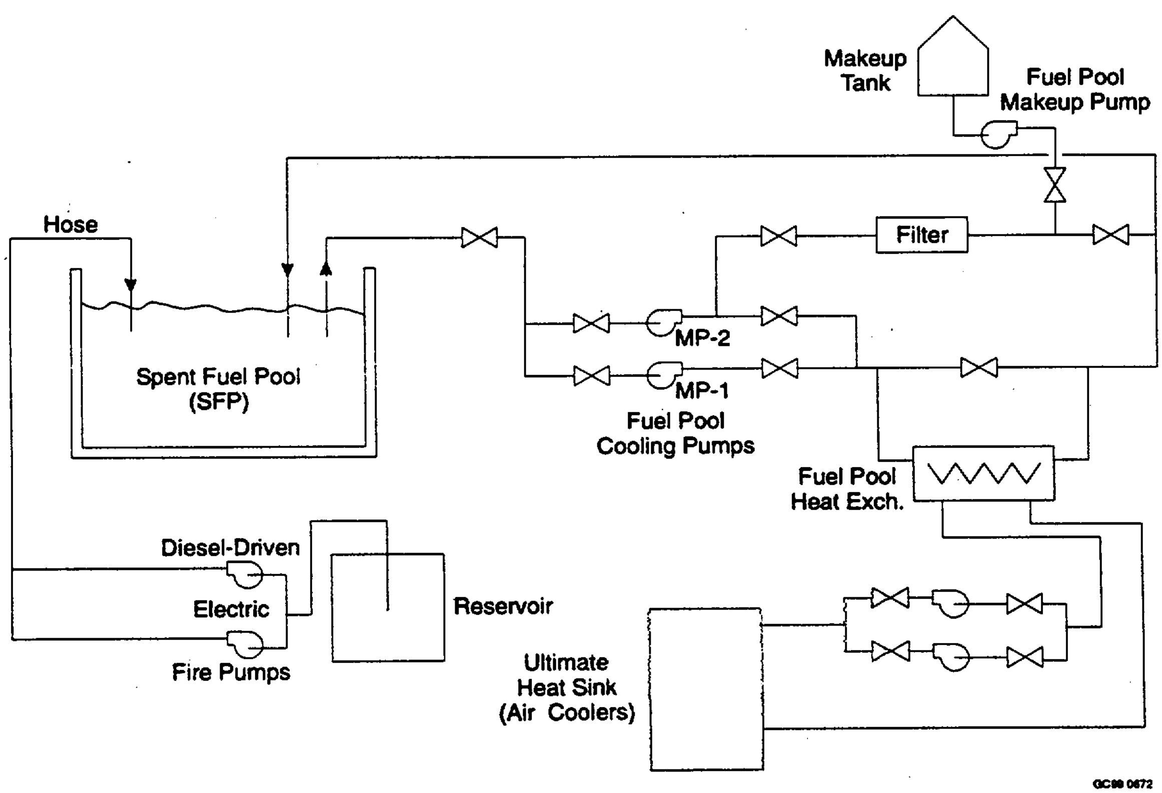
resources (public or private) to bear is fundamental for ensuring that the fuel assemblies remain cooled and a zirconium fire is prevented.

As part of its technical evaluations, the staff assembled a small panel of experts to identify the attributes necessary to achieving very high levels of human reliability for responding to potential accident scenarios in a decommissioning plant SFP. (These attributes and the human reliability analysis (HRA) methodology used are discussed in Section 3.2 of Appendix 2A.)

Upon considering the sensitivities identified in the staff's preliminary study and to reflect actual operating practices at decommissioning facilities, the nuclear industry, through NEI, made important commitments, which are reflected in the staff's updated risk assessment.

Additional important operational and design assumptions made by the staff in the risk estimates developed in this study are designated as SDAs and are discussed in later sections of this study.

Figure 3.1 Assumed Spent Fuel Pool Cooling System



Industry Decommissioning Commitments (IDCs)

IDC #1	Cask drop analyses will be performed or single failure-proof cranes will be in use for handling of heavy loads (i.e., phase II of NUREG-0612 will be implemented).
IDC #2	Procedures and training of personnel will be in place to ensure that onsite and offsite resources can be brought to bear during an event.
IDC #3	Procedures will be in place to establish communication between onsite and offsite organizations during severe weather and seismic events.
IDC #4	An offsite resource plan will be developed which will include access to portable pumps and emergency power to supplement onsite resources. The plan wouldprincipally identify organizations or suppliers where offsite resources could be obtained in a timely manner.
IDC #5	Spent fuel pool instrumentation will include readouts and alarms in the control room (or where personnel are stationed) for spent fuel pool temperature, water level, and area radiation levels.
IDC #6	Spent fuel pool seals that could cause leakage leading to fuel uncovery in the event of seal failure shall be self limiting to leakage or otherwise engineered so that drainage cannot occur.
IDC #7	Procedures or administrative controls to reduce the likelihood of rapid draindown events will include (1) prohibitions on the use of pumps that lack adequate siphon protection or (2) controls for pump suction and discharge points. The functionality of anti-siphon devices will be periodically verified.
IDC #8	An onsite restoration plan will be in place to provide repair of the spent fuel pool cooling systems or to provide access for makeup water to the spent fuel pool. The plan will provide for remote alignment of the makeup source to the spent fuel pool without requiring entry to the refuel floor.
IDC #9	Procedures will be in place to control spent fuel pool operations that have the potential to rapidly decrease spent fuel pool inventory. These administrative controls may require additional operations or management review, management physical presence for designated operations or administrative limitations such as restrictions on heavy load movements.
IDC #10	Routine testing of the alternative fuel pool makeup system components will be performed and administrative controls for equipment out of service will be implemented to provide added assurance that the components would be available, if needed.

Staff Decommissioning Assumptions (SDAs)

SDA #1 Licensee's SFP cooling design will be at least as capable as that assumed in the risk assessment, including instrumentation. Licensees will have at least one motor-driven and one diesel-driven fire pump capable of delivering inventory to the SFP.

SDA # 2 Walk-downs of SFP systems will be performed at least once per shift by the operators. Procedures will be developed for and employed by the operators to provide guidance on the capability and availability of onsite and offsite inventory makeup sources and time available to initiate these sources for various loss of cooling or inventory events.

SDA # 3 Control room instrumentation that monitors SFP temperature and water level will directly measure the parameters involved. Level instrumentation will provide alarms at levels associated with calling in offsite resources and with declaring a general emergency.

SDA # 4 Licensee determines that there are no drain paths in the SFP that could lower the pool level (by draining, suction, or pumping) more than 15 feet below the normal pool operating level and that licensee must initiate recovery using offsite sources.

SDA # 5 Load Drop consequence analyses will be performed for facilities with non-single failure-proof systems. The analyses and any mitigative actions necessary to preclude catastrophic damage to the SFP that would lead to a rapid pool draining would be sufficient to demonstrate that there is high confidence in the facilities ability to withstand a heavy load drop.

SDA # 6 Each decommissioning plant will successfully complete the seismic checklist provided in Appendix2B to this study. If the checklist cannot be successfully completed, the decommissioning plant will perform a plant specific seismic risk assessment of the SFP and demonstrate that SFP seismically induced structural failure and rapid loss of inventory is less than the generic bounding estimates provided in this study ($<1x10^{-5}$ per year including non-seismic events).

SDA # 7 Licensees will maintain a program to provide surveillance and monitoring of Boraflex in high-density spent fuel racks until such time as spent fuel is no longer stored in these high-density racks.

Seismic Hazard Issues:

1) Seismic Hazard Curve Outlier - HB Robinson Plant:

Robinson was the highest eastern and central US seismic hazard using LLNL methodology; not highest using EPRI methodology

<u>LLNL vs EPRI hazard curves</u>:

The LLNL 1993 seismic hazard for the site is about a factor 100 higher than the EPRI 1989 hazard estimate. The original LLNL 1989 seismic hazard estimate was updated to account for "known errors" in the treatment of uncertainty in the magnitude frequency relationship and to accommodate some refinements in the expert opinion elicitation process. However, the seismic source zones were not changed because of cost impact.

<u>Charleston earthquake effect on HB Robinson:</u>
One unique feature of the Robinson site is the proximity to the Charleston event in South Carolina - 125 miles. It appears that the LLNL experts for seismic source were more influenced by the Charleston event than were the EPRI experts. Both seismic hazard estimates are considered credible.

More recent (90s) geological investigations of paleo-liquefaction features led to the conclusion that the Charleston event is confined to the Charleston area. A new seismic source characterization for this site can produce a different result, but the USGS seismic hazard map, a more recent (mid to late 90s) work, shows fairly high seismic hazard values for this site.

<u>NRC-sponsored study on seismic failure of SFP that included Robinson:</u>

NUREG/CR-5176 "Seismic Failure and Cask Drop Analysis of the Spent Fuel Pools at Two Representative Nuclear Power Plants," January 1989, was performed by LLNL in support of GI-82. LLNL found that the Robinson SFP is capable of handing a 0.65 pga (peak ground acceleration) earthquake

TWG study generically assumed at earthquakes at 0.5 pga would damage pool, therefore, Robinson could have site-specific justification for seismic events

2) IPEEE: DID NOT EVALUATE SPENT FUEL POOL

Cask Drop Events:

GI-82:
1) assessed probability of drop estimated to be < E-8 / reactor-yr in NUREG/CR-4982, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82," July 1987

2) NUREG/CR-5176, "Seismic Failure and Cask Drop Analysis of the Spent Fuel Pools at Two Representative Nuclear Power Plants" describes the FEM analysis of Robinson and Vermont Yankee plants. The drop was considered <u>on the side walls</u> from <u>heights of 4 -6 inches</u>. The model considered cask drops on top of walls that are approximately 4.5 ft thick by 40 ft deep acting as a deep beam spanning about 55 ft. Various cask designs (largest 110 tons) were used. The result is extensive damage to concrete and large areas of reinforcement yielding. Statement on Page 7-4, "...pool walls similar to those of both the Vermont Yankee and Robinson plants would suffer severe damage as a result of the worst-case cask drops."

The NUREG concluded:
- pool walls would suffer severe damage
- integrity of liner would be difficult to predict, but likely that the liner would be severely damaged
- loss of pool water could not be ruled out

TWG:

1) The staff assessed probability of cask drop over <u>complete load path</u>, including <u>the wall and SFP floor</u> to be 2.0 E-7

2) Cask drops inside the pool/on the floor:
There is no analysis of this case in NUREG/CR-5176. If the pool floor slab is assumed to be supported on unyielding foundation, the cask would probably not go through the floor, but cause local impact zone failure - considerable water leaking into the foundation. If the slab is not directly supported by the foundation, a 110 ton cask dropped from 4 ft height would go through the slab based on very approximate energy balance. The potential energy of the 110 ton drop from 4 ft height is 0.9 million ft-lbs, but the available resistance energy (calculated as work done by the punching shear in 4.5 ft thick concrete slab on a generous shear area and high shear stress) is about a tenth of that required. The travel of the cask though water before impact will hardly dissipate any energy - in the order of 1 to 5% or less. The claim that the cask will not go through a typical spent fuel pool slab not supported by the foundation will have to be validated by credible analysis.

Spent fuel pool design

	Plant	Date of Last Shutdown	Comment
1	Vallecitos (VBWR)	Dec. 9, 1963	No fuel on site
2	Saxton	1972	No fuel on site
3	Fermi 1[a]	Sep. 22, 1972	No fuel on site
4	Peach Bottom 1[a]	Oct. 31, 1974	No fuel on site
5	Indian Point 1	Oct. 31, 1974	
6	Humbolt Bay	Jul. 2, 1976	Below water table
7	Dresden 1	Oct. 31, 1978	outside containment on bedrock
8	La Crosse	Apr. 30, 1987	in containment stainless steel fuel
9	TMI 2	Mar. 28, 1979	
10	Rancho Seco	Jun. 7, 1989	soil foundaion
11	Yankee Rowe	Oct. 1, 1991	2 tiered fuel on bedrock
12	Trojan	Nov. 9, 1992	pool above grade SC1 pool don't know soil or bedrock
13	San Onofre 1	Nov. 30, 1992	fuel below grade (bedrock) going to ISFSI
14	Millstone 1	Nov. 4, 1995	pool in reactor building (not in containment) pool above grade
15	Haddam Neck	Jul. 22, 1996	5 - 6 ft of fuel is above grade (bedrock)
16	Maine Yankee	Dec. 6, 1996	SFP in bedrock fuel below grade
17	Zion 2	Sep. 19, 1996	fuel below grade SC I pool don't know if bedrock or soil
18	Zion 1	Feb. 21, 1997	fuel below grade SC I pool don't know if bedrock or soil
19	Big Rock Point	Aug. 29, 1997	Inside containment

Note: (a) NMSS has project management responsibility