NUCLEAR WASTE DISPOSAL **CRISIS**

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Chapter 8

Spent Fuel Risks

The NRC first evaluated the spent fuel risk in the Reactor Safety Study (RSS) released in October 1975.[1] The NRC had assumed that a spent fuel accident would only involve one-third of a reactor core's inventory, because the fuel assemblies discharged each refueling outage would be shipped offsite for reprocessing shortly thereafter. The NRC considered the spent fuel risk to be small compared to the risk from accidents involving the reactor core.

The National Environmental Policy Act of 1969 compelled the NRC to release an environmental impact statement for spent fuel storage in August 1979. The NRC reaffirmed its conviction that the "storage of spent fuel in water pools is a well established technology, and under the static conditions of storage represents a low environmental impact and low potential risk to the health and safety of the public."[2]

The NRC recognized that certain actions had eroded the basis for its original spent fuel risk analysis: after reprocessing was eliminated, utilities had expanded spent fuel storage capacities at nuclear power plants and disposal had been indefinitely deferred. The RSS had not considered so many spent fuel assemblies being stored for so many years. In addition, studies demonstrated that fire could propagate between irradiated fuel assemblies in the storage racks, a mechanism not contemplated in the RSS analysis. The NRC undertook a study in the early 1980s to determine if the interim spent fuel storage role presented unanalyzed accident scenarios or more severe consequences than previously analyzed. The study involved a probabilistic risk analysis of postulated spent fuel pool accidents initiated by random system failures, seismic events and dropping heavy loads. The analysis considered initiating event frequencies, system responses, and accident consequences such as cladding fires to evaluate the health effects from the postulated accidents.[3]

The NRC's study reported that a spent fuel pool accident involving fuel damage could result in an $8x10^6$ person-rem total radiation exposure to the 667,588 people living within a 50 mile radius of the plant. This radiological dose averages 11.98 Rem per person, equivalent to 479.2 times the maximum dose that federal regulations permit any member of the public to receive in an entire year. The study estimated that such an accident could result in off-site property damage totalling $3.4 billion in 1983 dollars. As in the RSS, the study assumed that the accident involved only the fuel discharged during the most recent refueling outage (i.e, one-third of a reactor core).[4]

However the NRC's study also reported that the chances of a spent fuel pool accident resulting in fuel damage were $1.5x10^{-7}$ per reactor year, or less than one accident every 60,000 years given the 109 plants currently operating. Due to the accident's perceived low probability, the NRC concluded that it represented an acceptable risk to public health and safety despite the severe consequences.[5]

The heart of probabilistic risk assessment (PRA) is statistical analysis. Such ciphering has valuable applications, but PRA proponents quantifying nuclear safety risks should consider the fact that a NRC statistician published this conclusion on March 9, 1979:

> The probability is less than 0.5 that the next (i.e., the first) major accident occurs within the next 400 reactor years. The probability is less than .05 that the next major accident occurs in the next 21 reactor years. The probability is larger than 0.5 that the next major accident will occur after the next 400 reactor years.[6]

The major accident at Three Mile Island Unit 2 occurred on March 28, 1979—fewer than 500 hours later.

The primary faults of PRAs include not addressing all credible initiating events and using invalid assumptions. It is exceedingly difficult to cover every conceivable failure mode and effect in a PRA for something as complex as a nuclear power plant. According to a consultant to the NRC who reviewed 25 Individual Plant Examinations featuring PRA, "attention to detail makes safe plants—lack of attention to details kills people."[7]

The nuclear power industry has not evaluated the integrated risk from nuclear power plant operation with the on-site storage of significantly more spent fuel assemblies than had been considered when the plants were designed. Spent fuel risk assessments assume that only one-third of a reactor core's inventory will be damaged, yet spent fuel pools now contain upwards of seven reactor cores of irradiated fuel assemblies as shown in Table 7-1. These details demand proper attention.

The spent fuel risk assessments dismiss the severe consequences from a spent fuel accident primarily due to the perceived long time that the operating staff has to perform mitigating actions. However, these risk assessments fail to account for the single most important element in any mitigation effort—namely, the problem's detection. The instrumentation used to monitor spent fuel pool temperature and level is almost always nonemergency equipment. This means that it is not designed, procured, installed, maintained, or tested with the same high standards applied to emergency system components to guarantee their performance. As repeatedly illustrated by the following incidents, the initiating event frequently goes undetected for hours or even days due to inoperable spent fuel pool instrumentation. It seems prudent, if not mandatory, to provide reasonable assurance that spent fuel pool problems will be readily detected before their grave consequences are dismissed based on remedial actions.

Loss of Water Inventory

The principal spent fuel accident concern is losing spent fuel pool water and the capability to cool the irradiated fuel assemblies. If the spent fuel pool drains, the spent fuel assemblies discharged within the past three to four years still produce sufficient decay heat to cause meltdown. In addition, the fuel's cladding could initiate and sustain rapid oxidation (often referred to as "fire" outside the nuclear power industry) during heatup prior to melting. The resulting cladding fire in a spent fuel pool equipped with high-density storage racks could spread to every spent fuel assembly.

The probability that the cladding would catch on fire after the spent fuel pool completely drains has been estimated at 100% for PWRs and 25% for BWRs.[8] The BWR probability is significantly lower because it was assumed that the BWR spent fuel assemblies are stored with their fuel channels in place, thus acting as barriers preventing the fire from spreading. Storing BWR spent fuel assemblies with the fuel channels in place significantly reduces spent fuel risk, yet the NRC does not require or even recommend that BWR plants implement this inexpensive safety precaution.

The loss of spent fuel pool water inventory event has the potential for contaminating the environment worse than would occur from a reactor core accident due to the significantly larger quantity of radioactive material available for release.[9] Additionally, the loss of spent fuel pool water inventory event is inherently worse than the reactor core accident because the fuel damage and radioactivity release occur *outside* the major barrier protecting the public, the primary containment. Therefore, it is more likely that radioactive material released in a spent fuel pool accident would reach the environment.

Several failure modes causing spent fuel pool water inventory to be lost were considered during the design process. The predominant failure mode is structural integrity damage that drains the spent fuel pool water at a rate exceeding makeup capability. The events producing this failure mode include earthquakes, heavy loads dropping into the pool or onto its wall, and turbine generated missiles. The secondary failure mode involves fuel pool cooling system malfunctions enabling accelerated water loss from the pool. The events producing this failure mode include a fuel pool cooling system pipe break and a failure of the system's heat removal function. Another failure mode, typically not considered during the design process but proving to be rather troublesome nonetheless, involves seal failure that allows water to leak from the pool into adjacent areas such as the containment, the shipping cask pit, and the fuel transfer tube.

The spent fuel pools at nuclear power plants in the United States are designed to withstand earthquakes without loss of integrity. The NRC evaluated the spent fuel pools at the Vermont Yankee and the H. B. Robinson plants to determine their vulnerability to earthquakes more severe than considered during design. They concluded that the spent fuel pools would probably survive an earthquake three times larger than they were designed to handle. They also concluded that it would take an earthquake nearly ten times greater than the design basis earthquake to cause the spent fuel pools to fail catastrophically.[10]

Spent fuel pools are not designed to withstand a shipping cask weighing 75 to 110 tons dropping onto their floors or walls. A dropped cask will probably cause the spent fuel pool to fail catastrophically. Although the consequences from a cask dropping into the spent fuel pool are significant, the probability that such an event will occur has been considered to be sufficiently low as to effectively manage this risk factor.

While the nuclear power industry has not experienced the prototypical cask drop event, there have been precursors. On December 28, 1994, a core shroud head bolt dropped into the Unit 1 spent fuel pool at Georgia Power Company's Edwin I. Hatch Nuclear Plant from one foot above the water surface when the sling holding the bolt broke. The bolt, 17 feet long by three inches in diameter and weighing 365 pounds, glanced off the side wall and fell to the bottom of the spent fuel pool without hitting the storage racks or irradiated fuel assemblies. The bolt tore a three inch gash in the 3/16 inch thick stainless steel liner. Approximately 2,000 gallons leaked through the hole and through a drain line to the radwaste system before valves in the drain line were manually closed. The SFP level dropped nearly two inches in 23 minutes, causing the fuel pool cooling system pumps to trip on low suction pressure. Operators restored level after the leakage path was isolated, then returned the fuel pool cooling system to service. Georgia Power removed the bolt and placed a large rubber mat (i.e., a nuclear-sized sink stopper) over the hole to limit leakage until underwater welding repairs were completed.

The Hatch incident occurred less than a year after a screwdriver dropped into the spent fuel pool at a foreign nuclear power plant with similar results. On January 31, 1994, workers at Tricastin Unit 1 in France were removing the control rod cluster guide tube from a spent fuel assembly. A 15 foot long screwdriver weighing 44 pounds fell into the spent fuel pool and punctured the stainless steel liner. The level in the spent fuel pool dropped nearly four inches. A stainless steel plate was welded over the hole.

Spent fuel pools are not designed to withstand the impact from a turbine generated missile. A turbine generated missile can result from the main turbine's gross failure. The detached blading or shroud from a large turbine spinning at 1,800 rpm can be extremely detrimental to whatever it impacts. The probability that a turbine generated missile will cause spent fuel pool integrity failure has been estimated to be $4.1x10^{-7}$ per reactor year. This probability is predicated on a $1x10^{-4}$ per reactor year probability that a turbine failure event generates a missile combined with a $4.1x10^{-3}$ probability that such a missile strikes the spent fuel pool with sufficient energy to be destructive.[11]

Following the main turbine failure at Fermi Unit 2 on Christmas Day, 1993, Detroit Edison Company determined that a high trajectory missile generated by the turbine could damage the spent fuel pool. The conditional probability of this occurrence, given the turbine failure, was estimated to be $1.0x10^{-4}$ per year.[12] As with the cask drop event, while the consequences from a turbine generated missile striking the spent fuel pool are significant, the probability that such an event would occur was considered to be sufficiently low as to effectively control this risk factor.

Spent fuel pools are designed to handle a loss of fuel pool cooling. This initiating event culminates in appreciable loss of spent fuel pool water inventory only when the spent fuel pool boils without makeup. This failure mode has been discounted in safety studies due to the extended period (relative to traditional reactor accident analysis time frames) available to restore cooling or provide makeup.

On January 25, 1994, Commonwealth Edison Company discovered considerable water in the basement of the containment structure at its Dresden Unit 1 plant. Dresden Unit 1 shutdown in October 1978 and remains virtually abandoned next to the operating Dresden Unit 2 and 3 plants. A service water system pipe in the unheated Unit 1 containment had frozen and ruptured, draining about 55,000 gallons from the system into the basement. Commonwealth Edison determined that piping in the spent fuel pool transfer system was also susceptible to freezing. If this piping had broken, the spent fuel pool would have drained to two feet below the top of the 660 irradiated fuel assemblies in the storage racks. At that level, the dose rate at the spent fuel pool railing was estimated at 733 Rem/hr, radiation levels that could have impaired operations on Dresden Units 2 and 3.[13] Dresden Unit 1 was not equipped with spent fuel pool level instrumentation to detect inventory loss.[14] This event had significant potential radiological consequences even though only 660 irradiated spent fuel assemblies resided in the spent fuel pool and these assemblies had undergone over 15 years of radioactive decay.

Failure of inflatable and mechanical seals is the most frequent reason that spent fuel pool water inventory is lost. Figure 8-1 illustrates various seal applications used in BWRs. Mechanical seals are used between the reactor pressure vessel and the containment structure (labeled "RPV to Drywell Bellows Seal" in Figure 8-1) and between the drywell and the refueling cavity (labelled "Drywell to Reactor Building Bellows" in Figure 8-1). Inflatable seals are used around removable gates (labeled "Gates" and "Double Gates" in Figure 8-1). Inflatable seals are like bicycle tire intertubes—when filled with air, they form a nearly leak tight barrier. The problem occurs when the inflatable seal loses air pressure and the barrier becomes rather porous.

The refueling cavity water mechanical seal (comparable to the "Drywell to Reactor Building Bellows" shown in Figure 8-1) at the Haddam Neck plant suffered a gross failure in August 1984 when mechanical interference significantly displaced the seal. At the time of the failure, the refueling cavity was flooded in preparation for refueling. The refueling cavity water level decreased 23 feet to the

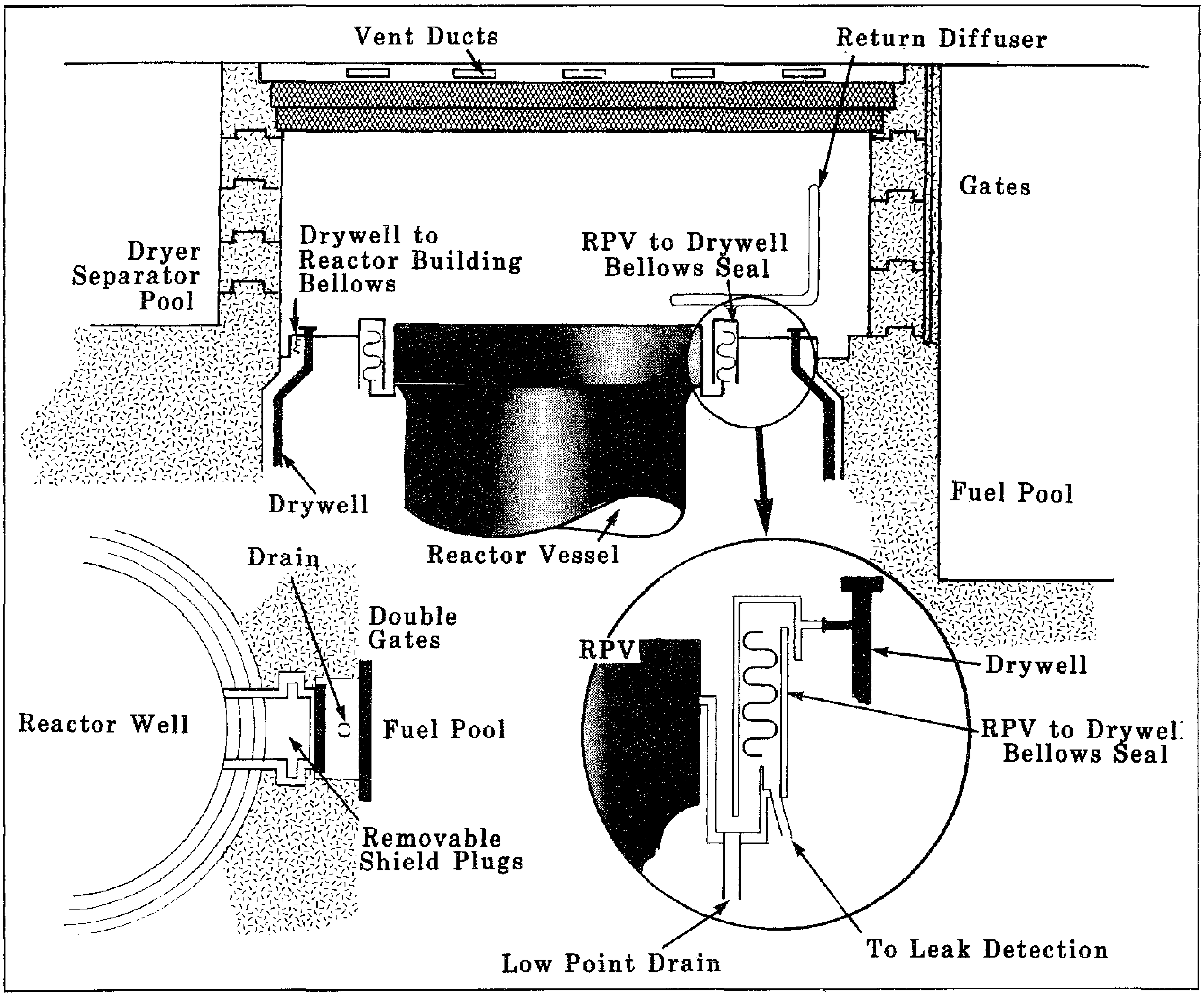

Figure 8-1 Reactor Well Seals

reactor vessel flange level within 20 minutes, flooding the containment with approximately 200,000 gallons. If a spent fuel assembly had been in transit at the time, it could have been partially or completely uncovered with potentially high radiation levels, fuel cladding failure and radioactivity release. In addition, if the fuel transfer tube had been open, the spent fuel pool could have drained to a level that would have uncovered the top of the irradiated fuel assemblies in the storage racks.[15]

The inflatable seal on the gate to the transfer canal between the Unit 1 and the Unit 2 spent fuel pools at the Edwin I. Hatch Nuclear Plant deflated in December 1986 after the air supply to the seals was mistakenly isolated. Nearly 141,000 gallons leaked from the spent fuel pools into the transfer canal, lowering the SFP level five feet. The leak was not identified for several hours because a leak detection instrument was inoperable at the time. Georgia Power determined that the leakage path could have drained the spent fuel pool to the bottom of the transfer canal, leaving only two feet of water over the top of irradiated fuel assemblies in the storage racks. The radiation field at the spent fuel pool railing would have been 100 Rem/hr in that condition, primarily from the control blades stored on the side of the spent fuel pool.[16] Several other incidents involving seal failure are described in Appendix A.

After the Haddam Neck event, the NRC required the postulated gross failure of the refueling cavity water seals to be evaluated for every nuclear power plant. The evaluation results varied due to different seal designs and refueling cavity geometries. Some plants required modifications to reduce the gross failure risk or provide seal leakage indication.

The results from the Northeast Utilities' evaluation of the Millstone Units 1, 2, and 3 plants for the Haddam Neck event represent typical findings. Northeast Utilities determined that in the unlikely event that the seal experienced catastrophic failure, the Millstone Unit 1 SFP level would drop to 20 inches above the irradiated fuel assemblies in 11 minutes with the resulting radiation field estimated to be 2.4×10^{6} Rem/hr at the spent fuel pool railing and 65 Rem/hr on the refueling floor. For the same postulated event on Millstone Unit 2, the SFP level would drop to 12 inches above the irradiated fuel assemblies in 80 minutes with the resulting radiation field estimated to be 4.0×10^{6} Rem/hr at the pool railing and 54 Rem/hr on the refueling floor. For the same postulated event on Millstone Unit 3, the SFP level would drop to 21 inches above the irradiated fuel assemblies in 120 minutes with the resulting radiation field estimated to be 1.9×10^{6} Rem/hr at the pool railing and 37 Rem/hr on the refueling floor.[17]

To put these radiation fields in perspective, a worker exposed to 37 Rem/hr receives the maximum annual radiation dose permitted by federal law in about 49 seconds, while a worker exposed to 1.9×10^{6} Rem/hr receives a fatal radiation dose in about one second. Because the probability that the refueling cavity water seal suffers catastrophic failure is considered to be negligibly small (despite already happening once), these potentially devastating consequences have been accepted by the NRC at Millstone and other nuclear power plants.

While spent fuel pool water inventory loss involves adverse consequences, too much water can also provide problems. On June 3, 1981, demineralized water leaked into the Surry Unit 1 spent fuel pool causing it to overflow. The spent fuel pool high level alarm was de-energized at the time and did not alert operators to the problem. Water overflowed into the new fuel storage area, then to the fuel receipt area and out the roll-up door to the station storm drains. In addition to the unmonitored radioactivity release, the demineralized water diluted the boron concentration in the spent fuel pool water and threatened subcriticality margin.

These spent fuel pool near-misses share many causal factors. In the majority of cases, the failure of a nonemergency system or component without the availability of a backup resulted in water inventory loss from the spent fuel pool. In many cases, the inventory loss was not promptly detected due to inoperable level instrumentation. The potential consequences from these events include high radiation fields and uncovering irradiated fuel assemblies outside primary containment. Given that federal regulations require the assumption that nonemergency systems and components fail or are unavailable following design basis events, the frequency of these spent fuel pool seal failures should warrant heightened attention, especially as more and more irradiated fuel assemblies are placed into the spent fuel pools.

Loss of Cooling

The inability of the fuel pool cooling system or its backups to remove the decay heat from the spent fuel pool poses an indirect challenge to public health and safety. The water in the spent fuel pool serves as a heat sink that provides time to restore fuel pool cooling prior to the spent fuel pool boiling. The boiling spent fuel pool produces a slow, sustained water inventory loss from the pool. As long as the boiling spent fuel pool's water level is maintained above the top of the irradiated fuel assemblies, these assemblies will be adequately cooled. However, the consequences from spent fuel pool boiling on the remainder of the plant can be extremely adverse as discussed in Chapter 9.

Fuel pool cooling can be lost for a large number of causes. Several incidents are described in Appendix A. As with the water inventory loss events, the primary cause for loss of spent fuel pool cooling events has been the failure of a non-emergency system or component. The loss of spent fuel pool cooling frequently remained undetected for several hours. The time that the spent fuel pool cooling failure goes undetected is important because federal regulations do not mandate a minimum time for the spent fuel pool water to reach boiling following loss of cooling, and it can require considerable time to restore fuel pool cooling. Thus, the spent fuel pool water can potentially heat up while a loss of cooling event remains undetected such that the pool boils before cooling can be restored. The severe consequences from spent fuel pool boiling are described in Chapter 9.

Radiation Overexposure

The water in spent fuel pools serves as a cooling medium to remove decay heat and as a shielding mechanism against the highly radioactive spent fuel assemblies. The spent fuel pool water inventory loss events previously described involved the potential for radiation overexposure as the shielding provided by the water was eliminated, but the following events and similar events summarized in Appendix A involve potential and actual radiation overexposure that occurred even though the required spent fuel pool water level was maintained.

The health physics (HP) technicians at the Browns Ferry Nuclear Plant were instructed during the summer of 1980 not to use portable teletectors to obtain contact dose rates for spent fuel assemblies. Portable teletectors are hand-held radiation monitors with a probe mounted at one end of a long aluminum tube and the readout at the other end. Apparently, HP technicians at other plants had dipped the probe into the spent fuel pool to determine the radioactivity level of spent fuel assemblies. The hollow aluminum tubes on the teletectors displaced the water, significantly reducing the shielding factor and providing a streaming pathway. The HP technicians standing at the spent fuel pool railing holding the teletectors at waist height pointed at spent fuel assemblies essentially had "guns" aimed at vital organs.

In June 1982, a diver installing storage rack support plates in the Indian Point Unit 2 spent fuel pool received an exposure of about 8.7 Rem to his head. A second diver received a whole body dose of about 1.6 Rem. An irradiated fuel assembly had been inadvertently transferred to a storage location two to four feet from the divers' work area. Limited visibility in the pool caused by cloudy water and insufficient underwater lighting prevented the misplaced fuel assembly from being detected. Alarming dosimeters mounted inside the divers' helmets failed to alarm at the 200 mr setpoint.[18]

While these events involved tangible hazards to individuals' well-being, the general public's health and safety was never at risk. Therefore, the risk of radiation overexposure when the spent fuel pool level is maintained is confined to nuclear power plant workers.

Handling Mishaps

The operation of over 100 nuclear power plants in this country since Shippingport's startup in December 1957 has required several hundred thousand irradiated fuel assembly movements. Spent fuel assemblies are discharged from the reactor core and irradiated fuel assemblies are repositioned within the reactor core every refueling outage. The entire reactor core is periodically off-loaded one assembly at a time to the spent fuel pool to allow in-vessel work, with fuel assemblies later reloaded for the next operating cycle. Spent fuel assemblies are moved within the spent fuel pool for inspections and to allow storage rack replacements. Equipment failures and personnel errors during these activities have resulted in a few hundred minor incidents such as the following events and similar events summarized in Appendix A.

During a refueling outage at Pilgrim in December 1979, an irradiated fuel assembly was inadvertently lifted from the storage racks high enough to activate the area radiation alarms on the refueling floor. The reactor building overhead crane was transferring new fuel assemblies from the inspection stand to the storage racks in the spent fuel pool. After a new fuel assembly was placed into a storage rack, the lifting hook caught between the lifting bail and the fuel channel on an adjacent irradiated fuel assembly. The operating staff failed to realize that the irradiated fuel assembly was being lifted from the spent fuel pool until the radiation alarms sounded. The operators quickly returned the irradiated fuel assembly to its storage location.

A top nozzle separated from a spent fuel assembly during transfer to the new high-density fuel storage racks at Prairie Island in December 1981. This fuel assembly had operated in the reactor for three cycles prior to being discharged in August 1978. It failed at a mechanical ball joint between stainless steel and zircaloy. The failure involved 16 joints in the area of maximum curvature and was caused by stress corrosion cracking of the stainless steel. The stress corrosion cracking was not believed to have occurred during reactor operation because the hydrogen overpressure should result in low free oxygen levels. However, the spent fuel pool has highly oxygenated water at low temperature that could cause cracking in the presence of high stresses and sensitized stainless steel. A total of 27 spent fuel assemblies at Prairie Island were examined with 12 showing signs of corrosion.[19]

At Brunswick in October 1994, while being seated on a loaded IF-303 spent fuel shipping cask, a closure head became cocked and required realignment. During the lifting operation, the closure head became stuck on one side of the cask causing two of four lifting cables to become overstressed and break. The reactor building overhead crane was not equipped with a load cell to indicate excessive loading.

The nuclear power industry encountered a number of fuel handling incidents in the early days that compelled hardware changes and training upgrades. Once the break-in period passed, the frequency of the fuel handling incidents decreased to approach that level defined by random equipment failures and personnel errors.

The risk from a fuel handling incident is limited to a single irradiated fuel assembly and whatever it strikes during transport or after being dropped. Nuclear power plants analyze postulated refueling accidents to ensure that the radiological consequences are within 10 CFR, Part 20 guidelines for nuclear power plant workers and 10 CFR, Part 100 guidelines for the public and the environment. The typical analysis assumes an irradiated fuel assembly is dropped from the maximum allowable height onto irradiated fuel assemblies in the reactor core or in the spent fuel pool's storage racks. The dropped assembly impacts one or more fuel assemblies before coming to rest. The number of fuel rods damaged in the event is conservatively determined from an energy balance and the fuel assemblies' material properties. The fission products released from the damaged fuel rods are estimated and applied to ventilation and filter configurations to develop the site and off-site radiological doses. The calculated dose to the public and to the workers is generally a small fraction of the allowable limits.

Criticality

Spent fuel storage racks are designed to maintain the spent fuel assemblies in a subcritical configuration. This passive function undertook increased significance when high-density storage racks featuring irradiated fuel assemblies in tightly packed arrays were introduced. A small number of incidents challenging this subcriticality requirement have been reported:

Gaps were measured in the Boraflex material used in the high-density fuel storage racks at Quad Cities Units 1 and 2 in May 1987. The gap formation mechanism was considered to be related to large local stresses in the Boraflex from fabrication-induced restraint within the rack and to tearing and shrinkage of the material. The average gap size was 1-1/2 inches, with the largest measuring nearly four inches. The gaps were discovered in the upper two-thirds of the cell length.[20]

During testing of selected South Texas Project Electric Generating Station Unit 1 storage racks in August 1994, 20 of the 37 storage cells tested showed some evidence of gaps or significant neutron absorption panel degradation. The neutron absorption panels contain a polymer matrix of boron carbide and silicon rubber. The large area degradations (up to 3 or 4-1/2 feet in length) represented accelerated dissolution of the Boraflex material caused by spent fuel pool water flowing through the poison enclosures. Samples of spent fuel pool water revealed increased silica amounts. Research determined that once the silicia concentration reaches equilibrium, the neutron absorption panel dissolution rate effectively stops. Removal of the silicia by the fuel pool cleanup system therefore sustains the degradation phenomenon. However, failure to remove the silica can adversely affect reactor coolant chemistry during refueling outages when the refueling cavity is directly connected to the spent fuel pool. Silica depositing on the fuel assemblies in the reactor core can adversely affect the heat transfer rate across the fuel rod cladding, causing fuel centerline temperatures to increase.

During the licensing process to obtain NRC's approval to install high-density storage racks, utilities frequently committed to test the adequacy of the neutron-absorbing material contained within the rack structure. This testing has led to the discovery of the deficiencies described above. Depending on the extent of the problem, the affected locations within a storage rack may be administratively barred from being used or the entire storage rack may be unloaded and replaced. As spent fuel pools are filled with irradiated fuel assemblies, these options may be restricted. For example, at some point a nuclear power plant may lose the capability to completely unload a storage rack due to the unavailability of sufficient empty locations in other racks.

Spent fuel pool criticality is managed exclusively through passive design features and administrative controls. Criticality in the spent fuel pool would yield significant consequences if it ever were to occur. First, nuclear power plants are not equipped with instrumentation to directly monitor subcriticality in the spent fuel pool. Second, nuclear power plants are not equipped with features to mitigate spent fuel pool criticality if it occurs, with the possible exception of PWRs that can borate the spent fuel pool water. And finally, nuclear power plants are not designed to shield against spent fuel pool criticality. The risk management is solely dependent on criticality prevention.

Passive Storage

The previous sections discussed the spent fuel risk from an initiating event such as the inadvertent drop of a fuel assembly or the failure of the fuel pool cooling system. The risk from spent fuel assemblies under normal storage conditions also has been evaluated. Understanding this passive risk is important because it applies to all nuclear power plants virtually all of the time; in fact, at all times except for those rare occasions when a fuel assembly is dropped or the cooling system fails.

Atomic Energy of Canada, Limited (AECL) and Ontario Hydro conducted the first major spent fuel pool storage study during 1977-1978. Several spent fuel assemblies from nuclear power plants and experimental reactors were disassembled and examined. The assemblies included in this study had been stored in spent fuel pools between 13 and 27 years. Destructive examinations were performed on specific fuel rods from each spent fuel assembly, while the remaining fuel rods were returned to the spent fuel pools for continued storage.

Ten years later, AECL and Ontario Hydro repeated the study to evaluate the effects after another decade of storage. AECL and Ontario Hydro visually examined the spent fuel assemblies. No indications of fuel rod cladding degradation (i.e., pitting, galvanic corrosion, etc.) were observed on any fuel assembly. Neutron radiography techniques were used to look for the presence of water inside fuel rods. If water was present inside a fuel rod, then that fuel rod's cladding was known to be defective. The neutron radiography examinations in 1988-1989 indicated that no fuel rods had failed since the examinations in 1977-1978.

During the 1988-1989 study, AECL performed gamma scanning of several spent fuel rods that had been in pool storage for 21 years. The gamma scanning results indicated no evidence of leaching or redistribution of fission products in fuel rods. Room temperature ring-tensile tests were performed on 30 fuel rods from 14 spent fuel assemblies that had been stored underwater for 13 to 27 years. The tests were conducted to determine if there had been any change in cladding mechanical properties due to storage under water. No significant difference in the cladding's mechanical properties were identified. AECL and Ontario Hydro concluded from the studies that spent fuel assemblies can be safely stored underwater for at least 50 years.[21]

EPRI and Battelle Pacific Northwest Laboratory researched empirical and analytical information available through 1984 on underwater spent fuel storage in this country. This team discovered no evidence that spent fuel assemblies degrade to any appreciable degree during long periods of underwater storage. The team concluded that substantial technical basis exists to justify allowing spent fuel to remain in wet storage for several decades.[22]

Managing the Risks

Nuclear power plants have stored spent fuel assemblies for nearly 40 years without any significant incidents. The primary threat to public health and safety from spent fuel storage results from loss of spent fuel pool water inventory. The consequences from a spent fuel pool accident include both radiation exposure (8×10^6 person-rem) and expense ($3.4 billion in 1983 dollars). Lesser threats to public welfare are posed by loss of spent fuel pool cooling, radiation overexposure to spent fuel, and spent fuel handling.

Near misses have been logged over the years, including numerous repetitions of the some events. Spent fuel risk assessments conclude that the probabilities of the worst case accidents occurring are reasonably low as to provide adequate assurance of public safety. However, these risk assessments are nonconservative because several of their key assumptions are invalid. For example, the risk assessments assume that a relatively long period is available following the initiating event to implement actions to prevent or mitigate resulting fuel damage. Yet the spent fuel pool temperature and level instrumentation at most nuclear power plants is nonemergency equipment with a history of frequently being inoperable or unavailable following initiating events. The initiating event may go undetected for several hours, thus reducing or eliminating the available response time.

The risk assessments nonconservatively assume that the operating staff has unlimited access to the systems that can be used to mitigate a spent fuel pool accident. The spent fuel pool, along with most of the equipment that cools and provides makeup water to the pool, are located inside the secondary containment structure at most BWR plants. Following a reactor accident involving fuel damage, secondary containment can be rendered inaccessible due to extremely high radiation levels. The consequences of such a reactor accident can mechanistically cause a loss of spent fuel pool cooling event. Hence, the risk assessments are wrong to assume unlimited access for restoration and mitigation efforts.

The risk assessments also nonconservatively assume that the probability of a fuel cladding fire propagating to adjacent fuel assemblies is significantly lower for BWRs than for PWRs because BWR fuel channels act as fire barriers. Many BWR spent fuel assemblies are stored with the fuel channels in place; not for risk reduction, but for convenience. There are no requirements or even recommendations that BWR spent fuel assemblies be stored with their fuel channels. Hence, it is wrong for the risk assessments to assume that the fuel channels are always present to prevent a fuel cladding fire in a BWR spent fuel pool from propagating.

The predominant invalid assumption in the spent fuel storage risk assessments involves the extent of fuel damage resulting from a spent fuel pool accident. From the initial NRC risk assessment in 1975 through the NRC's updated risk assessments in the 1980s, it has been assumed that only one-third of a reactor core's inventory would suffer damage following a spent fuel pool accident. Nuclear power plants currently store the equivalent of several reactor cores in closely packed arrays within the spent fuel pools (see Table 7-1). Studies have demonstrated that a spent fuel assembly discharged within the past three to four years still produces sufficient decay heat to cause its fuel cladding to burn. Studies have also demonstrated that the resulting cladding fire can spread to involve every irradiated fuel assembly in the spent fuel pool, even those that were discharged many years ago. Therefore, limiting the postulated releases from a spent fuel pool accident to that radioactivity contained in only one-third of a reactor core's inventory non-conservatively biases the consequences.

Risk is defined as the product of the probability that an event occurs and its consequences. Both the spent fuel pool accident's probability and its consequences have been underestimated. Therefore, the spent fuel risk is higher than perceived by the existing assessments.

The spent fuel risk is far from unmanageable. It only requires providing reasonable assurance that spent fuel assemblies remain covered with water at all times. The nuclear power industry has successfully kept spent fuel assemblies covered with water since Shippingport began commercial operation in 1957. The task grows increasingly more difficult as more and more spent fuel assemblies are stored at the nuclear power plants. To prevent the safety margin from diminishing, the nuclear power industry should place greater emphasis on the instrumentation monitoring spent fuel pool conditions. Spent fuel storage risk assessments should reflect actual plant configurations and operating conditions. These measures should provide reasonable assurance that spent fuel storage at nuclear power plants presents negligible risk to public health and safety.

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Chapter 9

The Susquehanna Concerns

The worst accident experienced by the American nuclear power industry occurred at Three Mile Island on March 28, 1979. Its Unit 2 reactor core suffered extensive damage. Radioactivity that escaped into the environment following this accident prompted government officials to recommend the evacuation of pregnant women and small children. But what if such an accident had occurred at a plant where nearly three times as many irradiated fuel assemblies were stored in its spent fuel pool as reside in its reactor core, and where the accident causes the fuel pool cooling system to fail and prevents operators from responding to this situation?

The answer is a disaster of Chernobyl proportions. Massive amounts of radioactivity would be released from fuel damaged in the reactor core and in the spent fuel pool, and the systems and structures designed to prevent this radioactive material from reaching the public would fail. The NRC was notified by a 10 CFR, Part 21 report dated November 27, 1992, that such a nuclear safety hazard existed in the design of the Susquehanna Steam Electric Station (SSES) and potentially existed in the design of 35 other operating nuclear power plants in the United States.[1]

A TMI-2 scale accident occurring at the Susquehanna plant on November 27, 1992, would have been catastrophic. The conditions resulting from the accident, including extremely high radiation levels inside containment, would have disabled or prevented the operation of systems required for reactor core cooling and makeup, spent fuel pool cooling and makeup, and filtration of gaseous effluents from containment. Without the cooling and makeup systems, the extent of reactor core damage may have exceeded that encountered at TMI-2; and the nearly 1,000 irradiated fuel assemblies in the spent fuel pool would also have been damaged.

The inability to restore cooling to the spent fuel pool at SSES would eventually lead to its boiling. Shortly after the pool started boiling, the containment would have failed. The radioactivity released from the damaged fuel in the reactor core and in the spent fuel pool would have proceeded uncontrolled and unfiltered to the atmosphere. The public health and safety would have been placed in great jeopardy.

The Susquehanna concerns are not science fiction. Following the Three Mile Island Unit 2 (TMI-2) accident, its containment could not be meaningfully entered for nearly a year due to the high radiation levels. Spent fuel pool cooling was never an issue following the TMI-2 accident for two reasons: first, the accident occurred three months after the plant began commercial operation and before its first refueling outage so its spent fuel pool did not contain any irradiated fuel assemblies; and second, the spent fuel pool at TMI-2 was located outside the containment building so access was not restricted by its high radiation levels.

Since the 10 CFR, Part 21 report to the NRC, PP&L has implemented numerous and costly design changes, procedure changes, and training program upgrades to significantly reduce the spent fuel risk at SSES. But what about the 35 operating nuclear power plants in this country with similar designs? Are they affected by the Susquehanna concerns? Do they need to implement all of the steps taken by PP&L to reduce the spent fuel risk? To date, the NRC has not bothered to ask owners of these plants to determine if their plants are equally vulnerable.

Description of the Concerns

Understanding the Susquehanna concerns can be intimidating due to the number of systems involved and their intricate relationships during a multitude of design basis events, but the concerns are simplified when viewed as a logically connected series of facts. These facts are depicted on decision trees (Figures 9-1 through 9-5) for the possible actions following the postulated initiating event—loss of fuel pool cooling on SSES Unit 1. The decision trees outline the questions that must be answered to provide reasonable assurance that the irradiated fuel assemblies in the spent fuel pool can be cooled under all design conditions without adversely affecting other safety systems.

The Susquehanna Steam Electric Station is a two unit nuclear power plant located adjacent to the Susquehanna River in Luzerne County, Pennsylvania, about 30 miles southwest of Wilkes-Barre and 10 miles north of Berwick. Each Susquehanna unit features a BWR from the General Electric Company's BWR/4 product line with a Mark II containment. Unit 1 began commercial operation in June 1983 while Unit 2 followed in February 1985. PP&L Resources (formerly the Pennsylvania Power & Light Company) is the primary owner and operates the facility.

Restoring SFP Cooling Using Unit 1 Systems

The postulated loss of fuel pool cooling on SSES Unit 1 is the entry condition in Figure 9-1. Fuel pool cooling can be lost for numerous reasons including an earthquake that breaks the fuel pool cooling system's piping, a loss of off-site power that interrupts the power supply to the system's pumps, and fuel pool inventory loss that trips the pumps on low net positive suction head (NPSH).

Box 9-1.1 questions accessibility to the Unit 1 reactor building. Accessibility is an important consideration following a loss of cooling event because Unit 1's

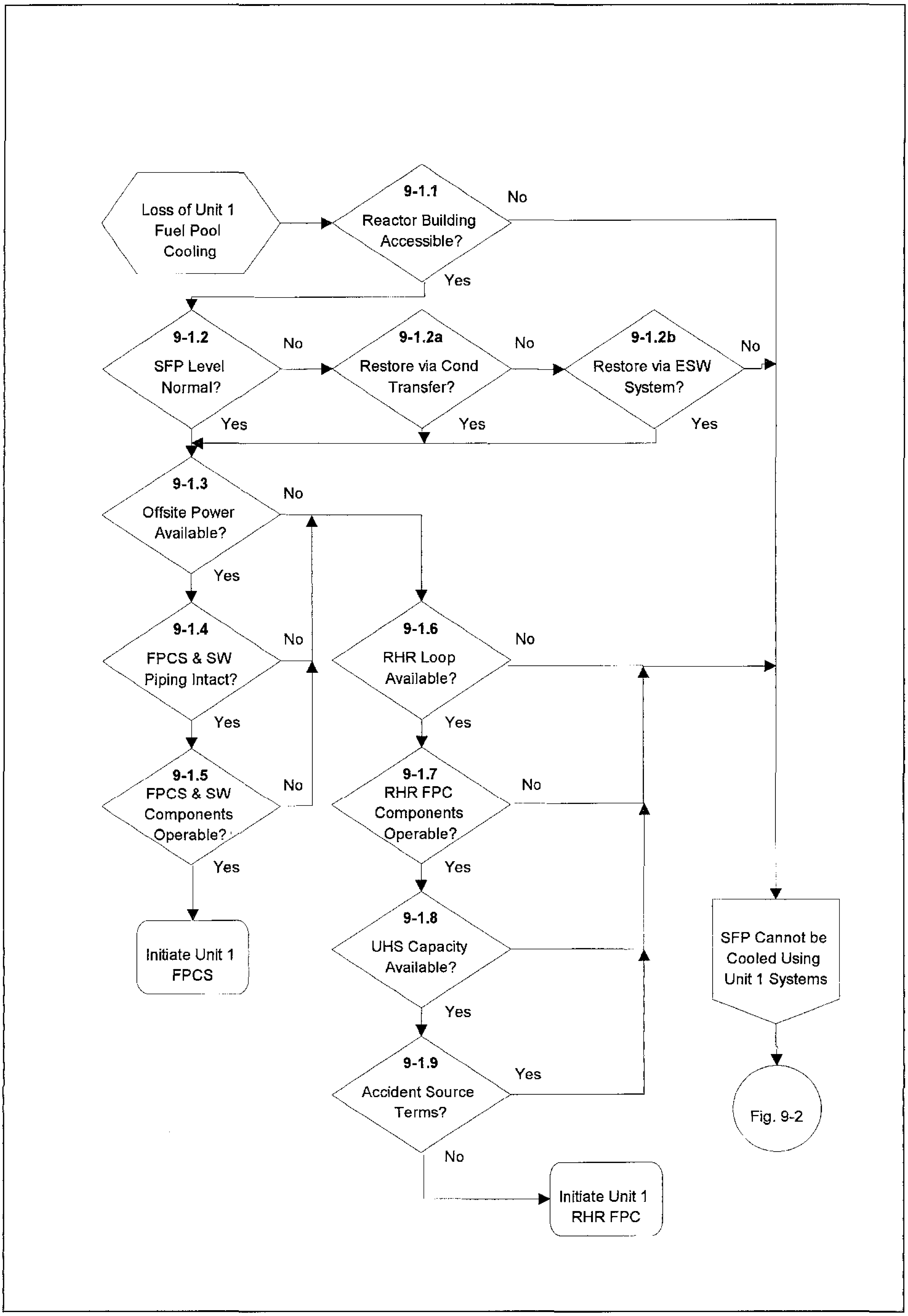

Figure 9-1 Restoring Spent Fuel Pool Cooling Using Unit 1's Systems

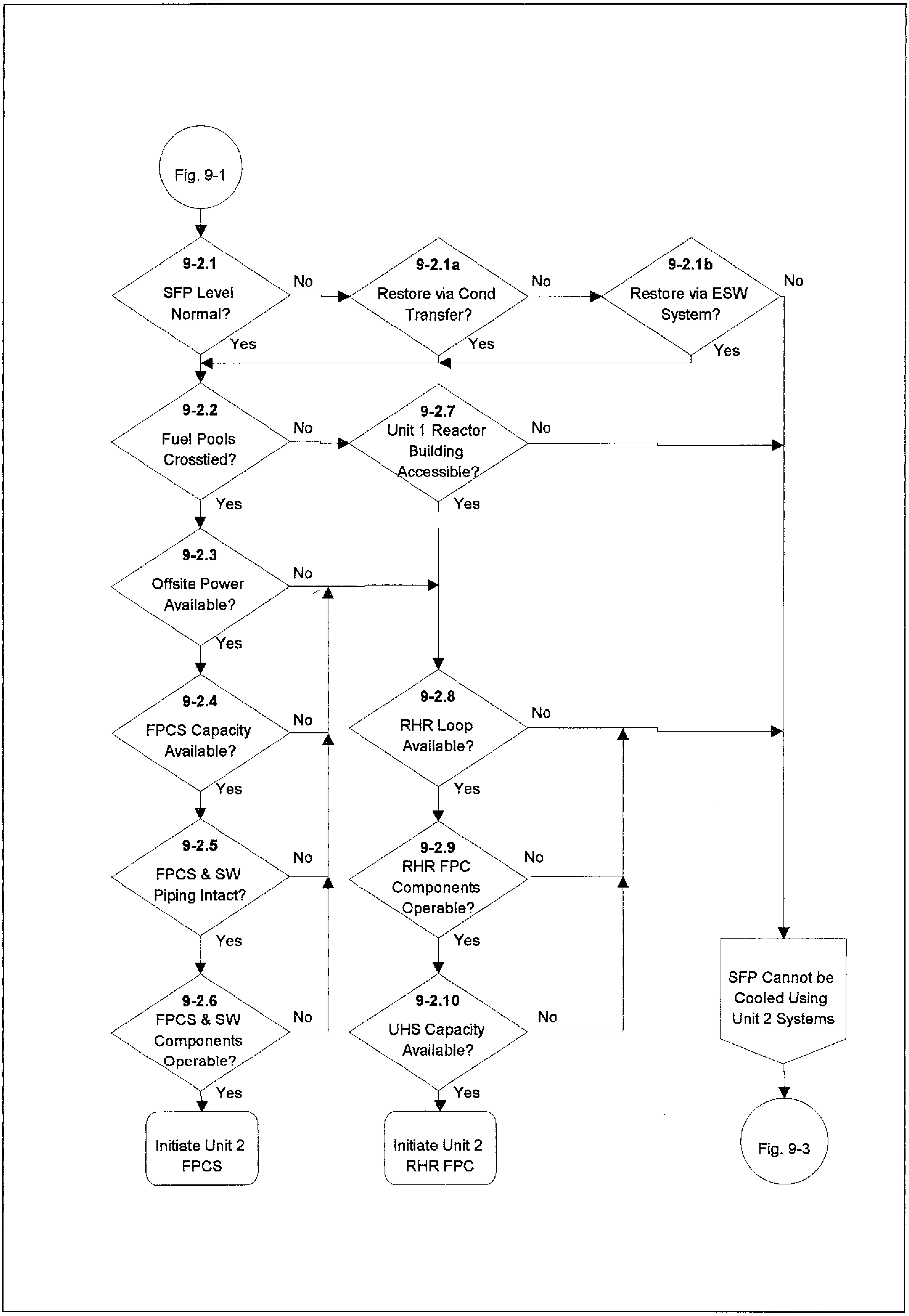

Figure 9-2 Restoring Spent Fuel Pool Cooling Using Unit 2's Systems

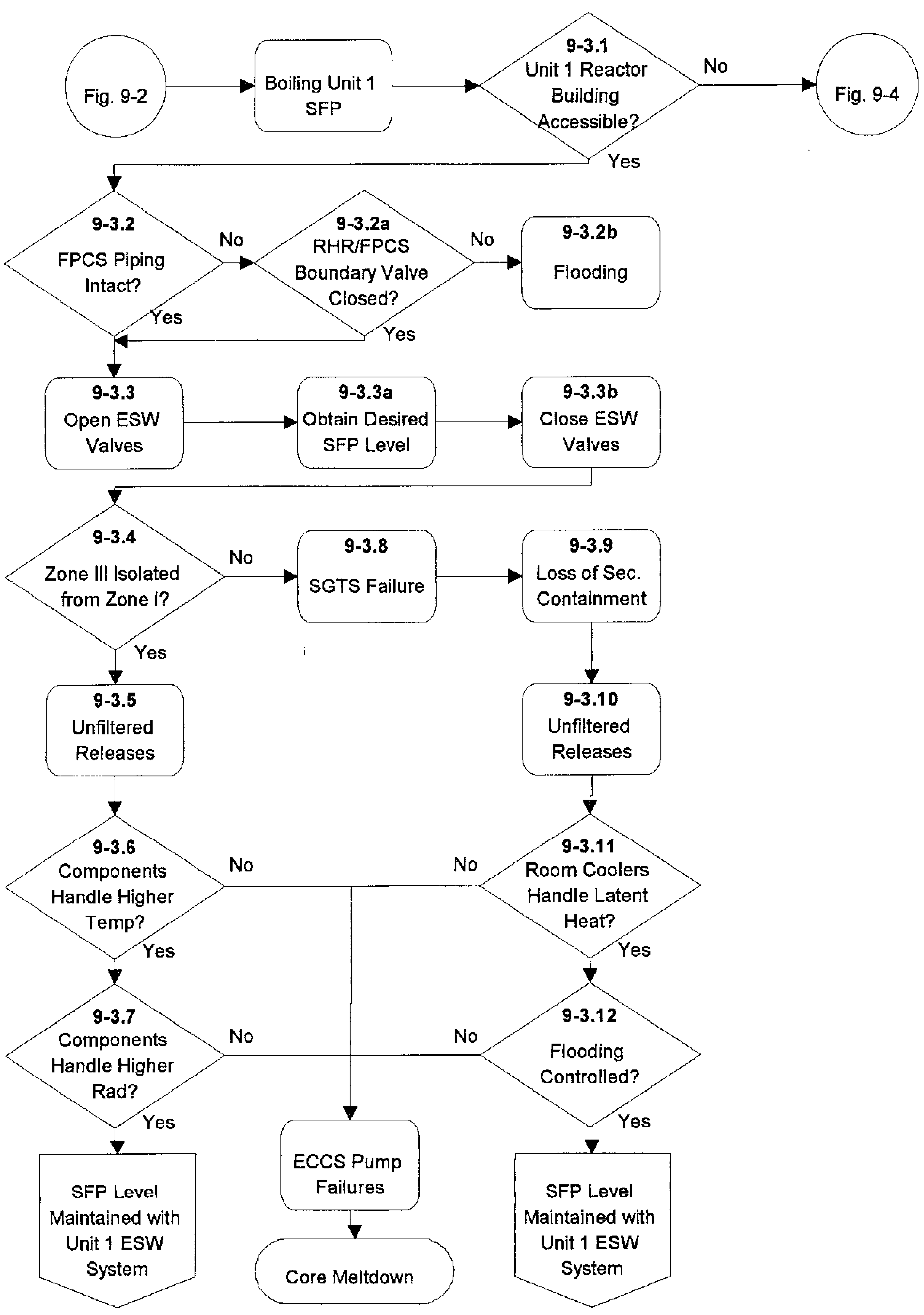

Figure 9-3 Handling Spent Fuel Pool Boiling Using Unit 1's Systems

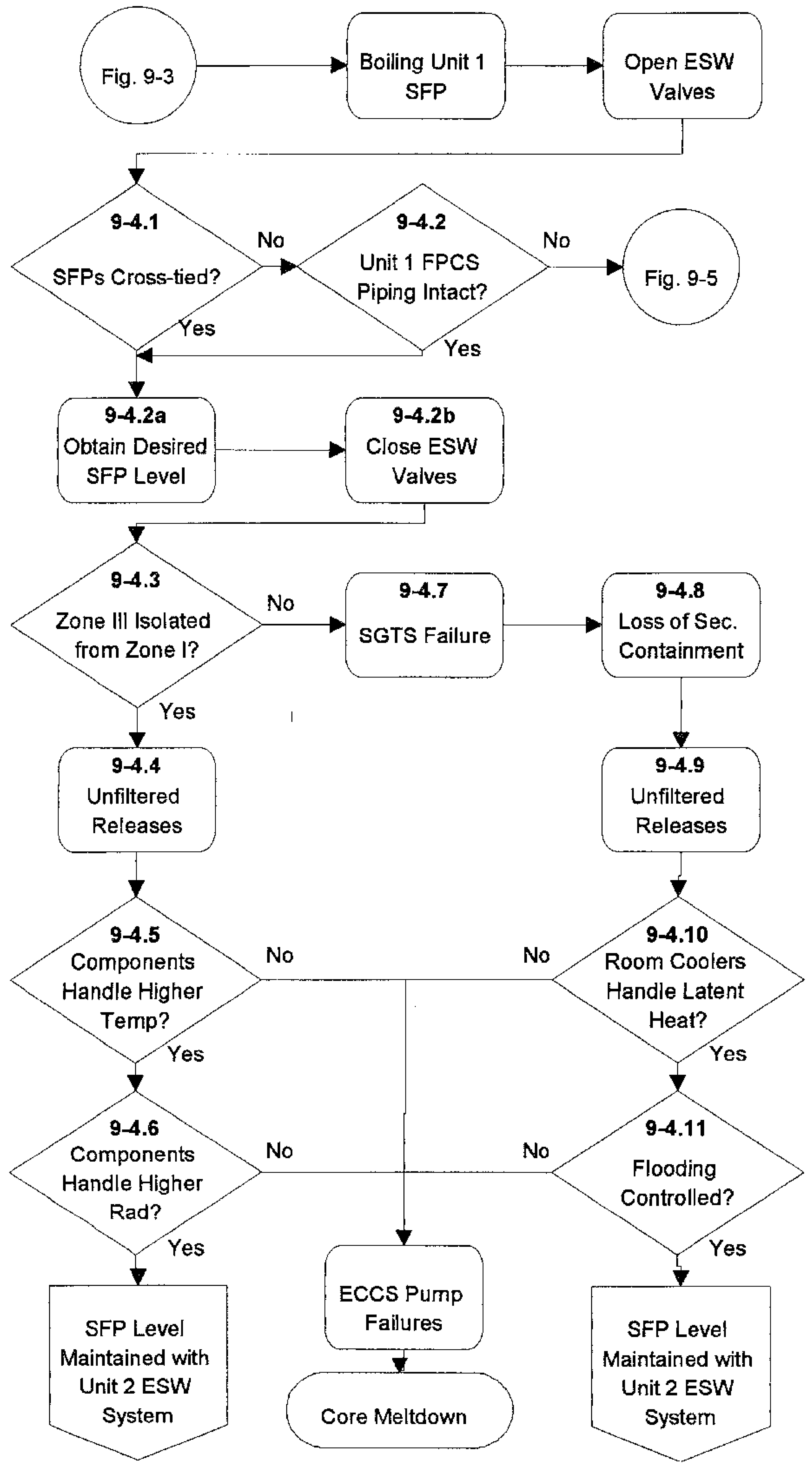

Figure 9-4 Handling Spent Fuel Pool Boiling Using Unit 2's Systems

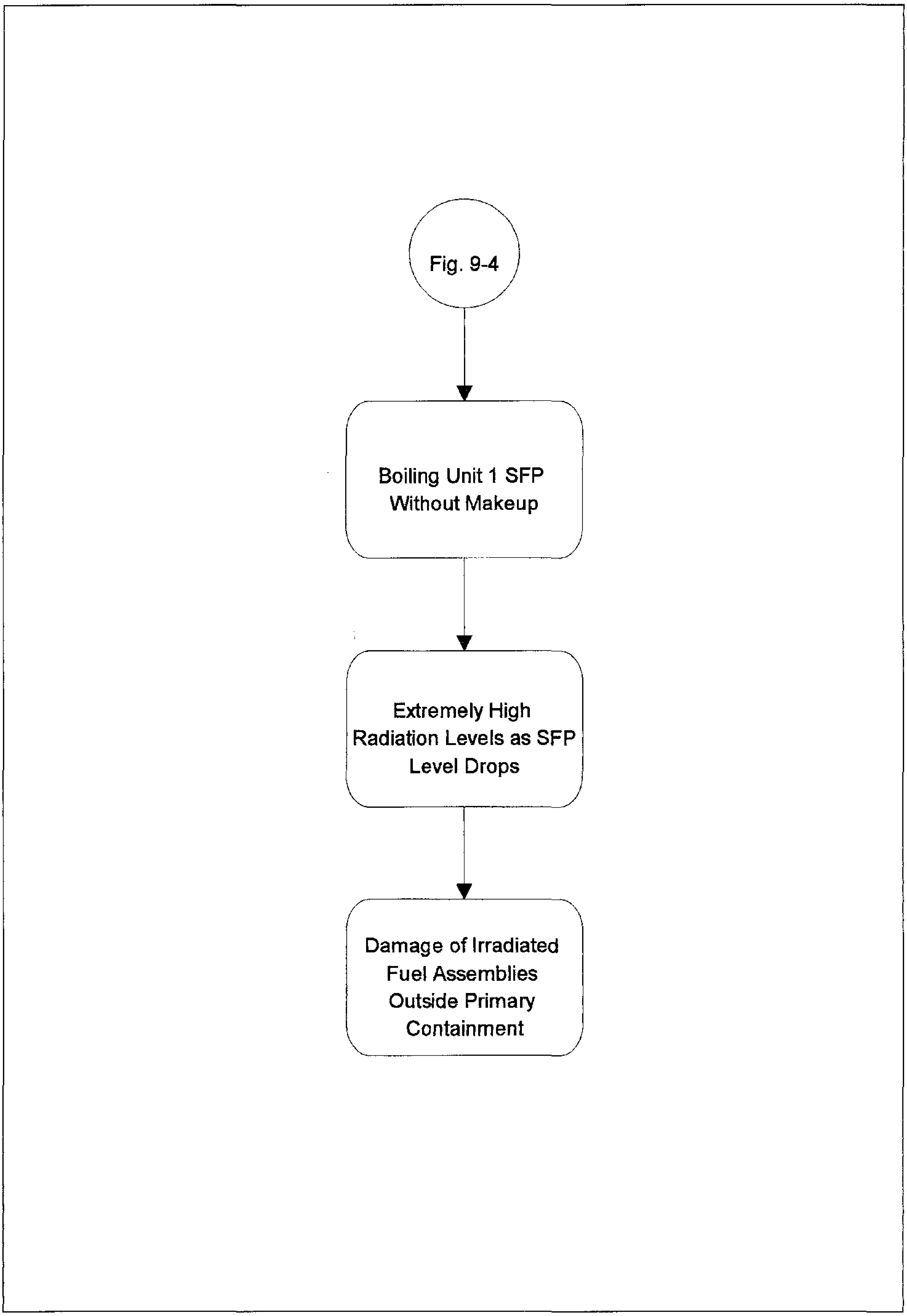

Figure 9-5 Spent Fuel Pool Boiling Ultimate Consequences

systems that either provide fuel pool cooling or provide makeup water to the fuel pool must be manually operated from within this structure.

Figure 9-6 illustrates a Mark II containment similar to the SSES design. The primary containment structure is the inverted cone-shaped structure housing the reactor pressure vessel. The secondary containment consists of the reactor building and the refueling floor. The only barrier preventing access to the reactor building would be environmental conditions hazardous to the operators—smoke, heat and toxic fumes from a fire, high temperature and high humidity from a pipe break, or high temperature and high radiation from a design basis loss-of-coolant accident (DBA LOCA).

The hazardous environment produced by a fire or by a pipe break may prohibit operator access, but only for a limited duration. Within an hour or a few hours at most, the fire will be extinguished or the pipe break will be isolated and the reactor building atmosphere will return to tolerable conditions. Operators could then enter the area to control the fuel pool cooling system or its backup. This minor accessibility inconvenience does not apply to the reactor building environment following a DBA LOCA.

The DBA LOCA is the maximum credible accident that nuclear power plants are required to experience without adversely affecting the public health and the environment. The NRC's positions on the DBA LOCA's role in assuring nuclear safety state:

> Design basis events (both transients and accidents) have been defined over the years and used to test the overall adequacy of each nuclear power plant design. These design basis events were intended to represent sound judgment regarding the reasonable range of events which might occur, and were thought to define a reasonable envelope of all credible events. Thus, the design of each plant was required to be capable of mitigating the consequences of those considered in the design basis.
>
> The most severe of this set of design basis events is the spectrum of loss of coolant accidents (LOCA). These accidents serve to set the requirements for a number of safety systems, including the emergency core cooling system (ECCS) and design of the containment building. In conjunction with these accidents, a coolant and fission product release into containment is assumed to occur, and these assumptions are used to set the leak-tightness of the containment and the capabilities of other engineering safety features.[2]
>
> For the purposes of this analysis, large fractions of the fission products were assumed to be released from the core even though these releases would be precluded by the performance of the ECCS.[3]

The important point is that nuclear power plants are required by federal regulations to protect the public by accommodating a postulated DBA LOCA with substantial fuel damage. The chances of such an event occurring are totally irrel-

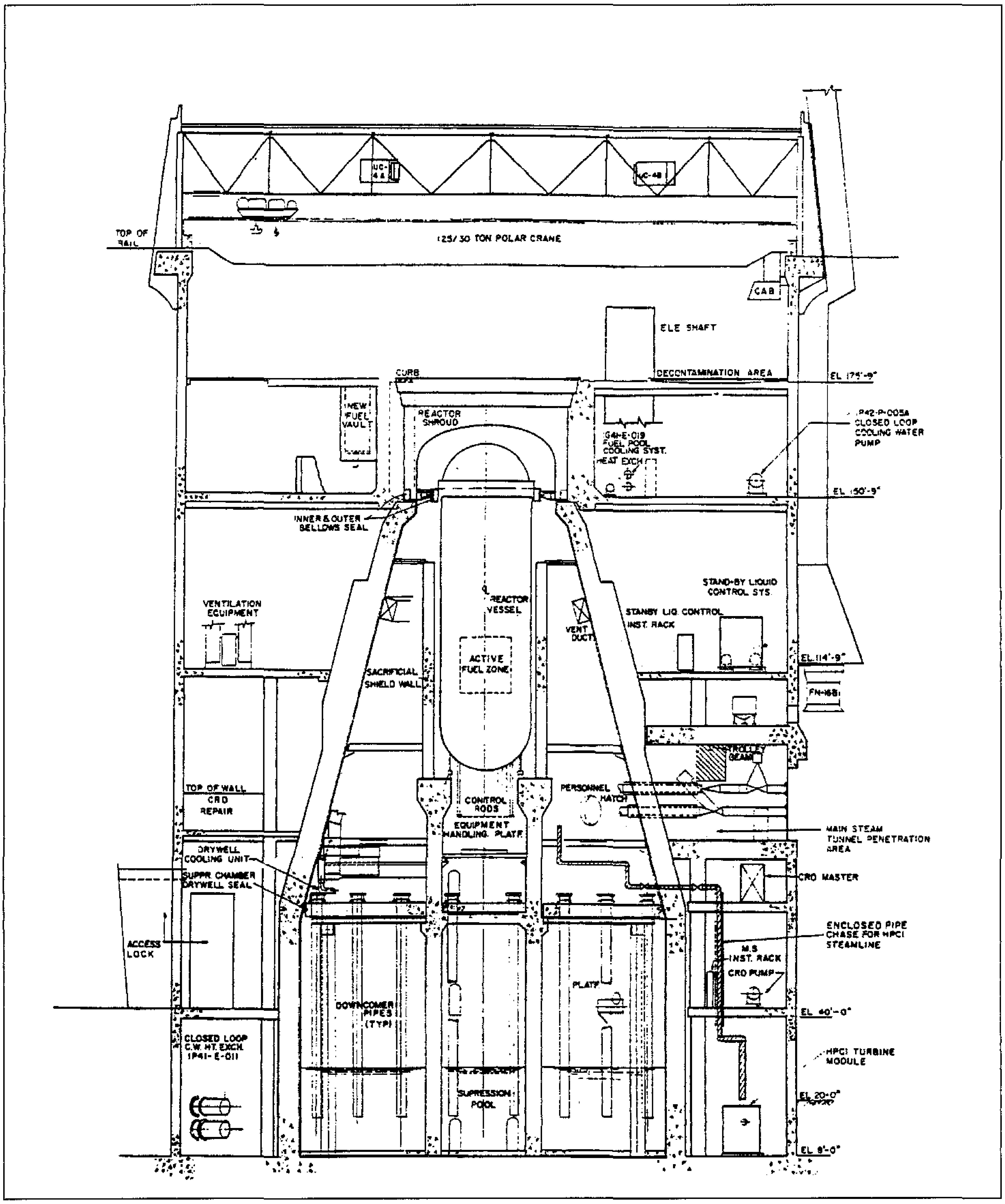

Figure 9-6 Mark II Containment Similar to the SSES Design

evant because in the design and operation of these plants, it is assumed to happen.

NRC Regulatory Guide 1.3 specifies the quantities of radioactive material that must be assumed to be released in a postulated design basis accident. PP&L used the Reg Guide 1.3 source terms in its analyses for SSES and concluded that "the reactor building will be generally inaccessible for several days after the accident."[4] PP&L calculated that the dose rates on the refueling floor peaked at 404.0 Rem/hr four days after an accident and dropped to 27.1 Rem/hr after 30 days.[5] The dose rates calculated in other reactor building areas were as high as 5,000 Rem/hr.

The lethal radiation dose is 450 to 600 Rem. Federal regulations limit the radiation exposure that an operator receives throughout the duration of a postulated design basis event to 5 Rem. In an area with a 5,000 Rem/hr dose rate, an operator reaches the federal exposure limit in under four seconds and receives a lethal dose in five to seven minutes. According to the radiation dose rates calculated by PP&L, the reactor building following a DBA LOCA with fuel damage will not be accessible enough to take any meaningful manual actions.

In 1985, the Office of Technology Assessment reported to the United States Congress that "if a TMI-type meltdown occurred at a boiling water reactor with a storage pool inside the containment, the operators would probably prefer not to have 20-30 years' worth of spent fuel stored in a place that might not be physically accessible for a period of years after the accident."[6] More recently, the Washington Power Public System changed the normal position of manual valves on the system inside the reactor building at its WNP-2 facility that provides makeup to the spent fuel pool "because LOCA could generate sufficient radiation fields in the Reactor Building to restrict accessing the manual valves prior to unacceptably high Spent Fuel Pool temperatures being reached."[7] Thus, it is generally accepted within the nuclear power industry that access to a BWR's reactor building following an accident with fuel damage will be limited, if not totally prohibited.

Fuel pool cooling cannot be restored using Unit 1's fuel pool cooling system or its only backup, the RHR fuel pool cooling assist mode, without access to the Unit 1 reactor building. The only event preventing this access is an accident with fuel damage. This event has a low probability of occurrence, but it is still a design requirement. The decision trees proceed to Figure 9-2 in this eventuality to determine if Unit 2's systems can be used.

When the Unit 1 reactor building is accessible, box 9-1.2 questions the Unit 1 spent fuel pool level. The fuel pool cooling system and its backup system require adequate pool level to operate. As shown in Figure 9-7, these systems take suction from the skimmer surge tank. After the spent fuel pool level drops below the point where water spills over into the skimmer surge tank, the pumps in both systems trip when the tank is pumped dry. In addition, the RHR fuel pool cooling assist mode requires the level to be raised eight inches above normal to provide sufficient return flow pathways to the skimmer surge tank to sustain the higher RHR pump flow rate.

The only impediment to adequate spent fuel pool level is inventory loss. Inventory is lost when the spent fuel pool liner integrity is breached or following uncompensated evaporation. With access to the Unit 1 reactor building, the level in the Unit 1 spent fuel pool can be restored and maintained using either the condensate transfer system or the emergency service water (ESW) system. In the unlikely combination of events required to cause fuel pool inventory loss and unavailability of both makeup systems, fuel pool cooling cannot be restored using Unit 1's systems. The decision trees proceed to Figure 9-2 in this eventuality to determine if Unit 2 systems can be used.

One design deficiency that was reported to the NRC concerned inadequate spent fuel pool instrumentation—a problem that indirectly affects the fuel pool level monitoring required to properly evaluate box 9-1.2. At the time of the 10 CFR, Part 21 report, fuel pool temperature and level indication at SSES was only provided on a local panel situated on the refueling floor. The operators in the control room had no fuel pool temperature or level indication.[8]

The fuel pool level indication at the local panel was actually inferred from the water level in the skimmer surge tank. When the fuel pool cooling system takes suction from the skimmer surge tank and returns water to the spent fuel pool where it overflows back into the skimmer surge tank, the tank's level represents the increase or decrease in spent fuel pool water inventory. However, following any event that fails fuel pool cooling system (FPCS) piping integrity, the skimmer surge tank instrumentation immediately becomes unavailable because the tank drains out through the break. This problem is relatively minor compared to the lack of indication in the control room since the operator dispatched to the local panel on the refueling floor who finds the level downscale could simply turn around and eyeball the spent fuel pool level.

The applicable federal regulation on spent fuel pool instrumentation states: "Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions."[9] The SSES design failed to comply with this regulation because the original instrumentation did not permit the operators to monitor spent fuel pool conditions and implement timely safety actions to either restore cooling or provide makeup. Following the 10 CFR, Part 21 report, PP&L installed temperature and level instruments in each unit's fuel pool with remote indication in the control rooms.[10]

Box 9-1.3 on Figure 9-1 questions off-site power availability. Off-site power availability is an important consideration because both the FPCS and the service water system (SWS) receive electrical power from non-emergency sources. Therefore, the pumps for these systems cannot operate if off-site power is lost.[11]

Box 9-1.4 questions FPCS and SWS piping integrity. Piping integrity is an important consideration in providing flow paths for cooling. Neither system is designed for the loadings resulting from a seismic event. Therefore, unless a seismic event has occurred, the system piping will probably be intact.

The operability of FPCS and SWS components is questioned in box 9.1-5. Component operability is an important consideration because a single compo-

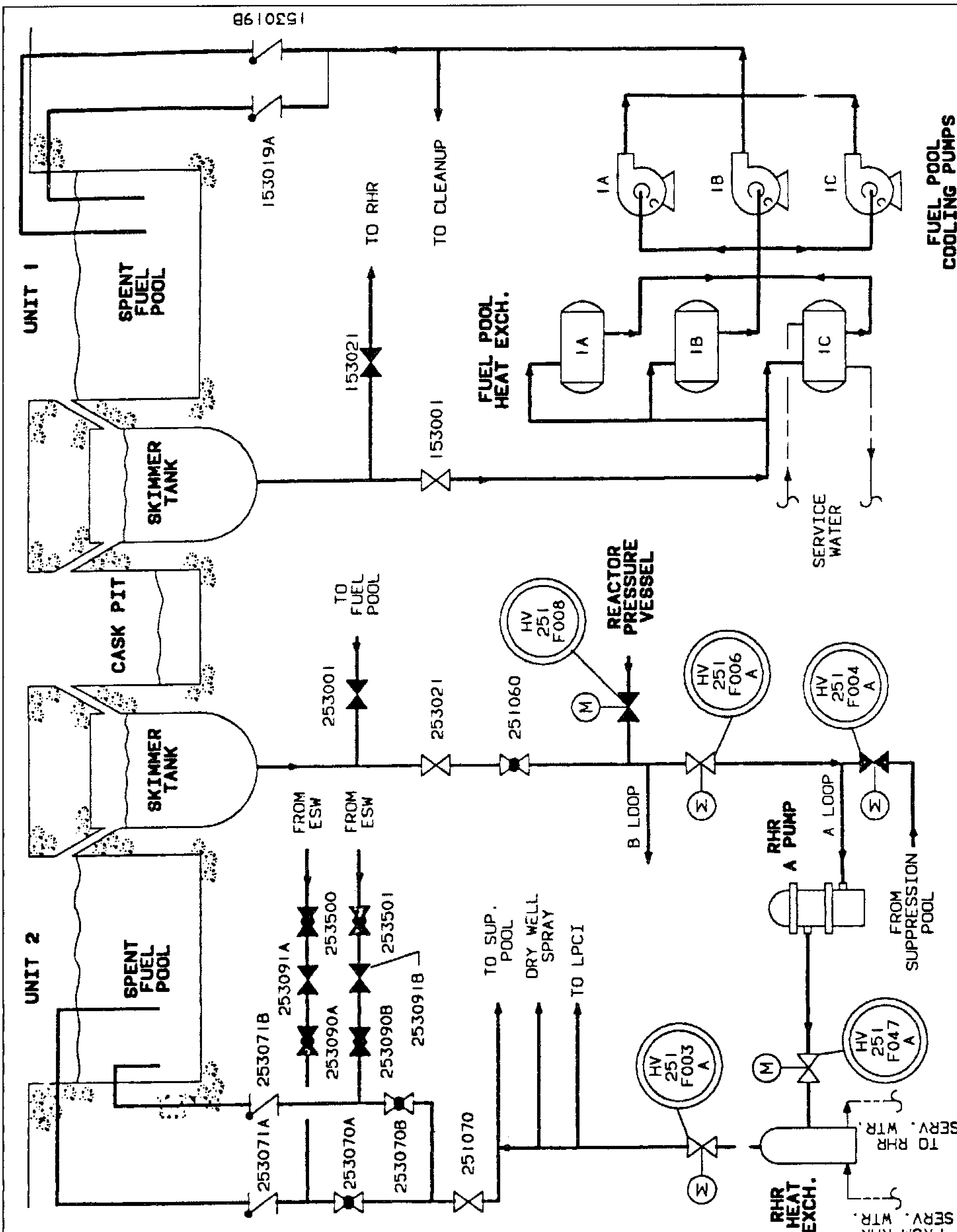

Figure 9-7 SSES Spent Fuel Pool Cooling Systems

nent failure in either system can disable fuel pool cooling capability. Federal regulations require that "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents."[12] The FPCS and SWS components located in the reactor building are not designed for the temperature, humidity, and radiation conditions that will be present following a DBA LOCA. These components may continue to operate beyond their design capabilities, but there is no assurance that they remain functional. Particularly vulnerable to failure under these conditions are the electrical components including pump motors, motor control centers, and control loops. Therefore, the FPCS and SWS components probably remain operable following events in which ambient conditions remain essentially unchanged, but one or more components will probably fail following events in which reactor building temperature, humidity, or radiation conditions increase substantially.

When the Unit 1 reactor building is accessible, the Unit 1 spent fuel pool level is normal, off-site power is available and the Unit 1 FPCS and SWS piping is intact with all required components operable, the Unit 1 FPCS can be used to cool the Unit 1 spent fuel pool. Unfortunately, that's the good news in a good news/bad news combination. The bad news is that loss of Unit 1 fuel pool cooling was the initiating event; therefore, one or more of these conditions is probably invalid.

When the Unit 1 FPCS cannot be used for fuel pool cooling, box 9-1.6 questions the availability of an RHR loop on Unit 1. The RHR system has numerous emergency functions that might impact the availability of a loop for the non-emergency RHR FPC assist function. For example, the RHR system provides makeup water for reactor core cooling following a DBA LOCA event and later provides suppression pool cooling. The RHR system in conjunction with the other low pressure ECCS systems are designed with redundancy so that necessary safety functions are achieved even with the failure of one RHR loop. However, SSES is not designed or analyzed to perform all necessary safety functions with the diversion of one RHR loop for FPC mode. Therefore, for many, but not all, design basis events at SSES an RHR loop will be available for RHR FPC.

When one RHR loop is available for FPC mode, box 9-1.7 questions the operability of the components required in this mode. Component operability is an important consideration because a single failure in this portion of the RHR system can disable it. The components involved are the manual valves (the Unit 1 equivalent of valves 253021, 251060, 251070, and either 253070A or 253070B in Figure 9-7) that must be opened to align the necessary flow path. PP&L removed these manual valves from the SSES Inservice Inspection and Testing Program in 1987 and 1988 after an engineering evaluation concluded that the RHR FPC function was not required. Following the 10 CFR, Part 21 report, PP&L resumed testing them after a five year lapse.[13] PP&L recently revised the FSAR to formally rely on RHR FPC to prevent spent fuel pool boiling following a seismic event.[14] Therefore, the RHR FPC components will now most likely be operable.

Box 9-1.8 questions the available capacity of the ultimate heat sink (UHS). The UHS at SSES is a large man-made cooling pond next to the plant. The RHR FPC mode transfers the decay heat it removes from the spent fuel pool to the RHR service water (RHRSW) system that in turn dissipates this energy to the atmosphere via the UHS. The RHRSW system takes water from the UHS, circulates it through the RHR heat exchangers, and returns the warmer water via two loops discharging through spray headers into the UHS pond where heat is removed by evaporative cooling. Following a DBA LOCA, the use of the RHR system in fuel pool cooling mode requires both UHS spray headers to maintain the UHS temperature below its maximum design limit. The design basis single failure in the UHS safety analysis is failure of one entire spray loop; thus, any alignment requiring both spray loops is beyond the design basis capability of the UHS.[15] Therefore, adequate UHS capacity is most likely available for all design basis events, with the significant exception that a single failure cannot be tolerated following a DBA LOCA.

The SSES design for the UHS violates federal regulations because it cannot handle the DBA LOCA event if a single failure is assumed. General Design Criterion (GDC) 44 of Appendix A to 10 CFR, Part 50 states:

> A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power operation (assuming offsite power is not available) and for offsite power system operation (assuming onsite power is not available) the system safety function can be accomplished assuming a single failure.

Per GDC 44, the Susquehanna plant shall be designed to transfer the heat load from the secondary containment to the UHS under accident conditions assuming off-site power is unavailable with a single failure. When off-site power is lost, the normal fuel pool cooling system cannot be used and RHR FPC must be used to cool the spent fuel pool. However, the UHS cannot handle the spent fuel decay heat loads and the other heat loads required to cope with the accident unless both spray headers are functional. A single failure cannot be tolerated; therefore, the federal nuclear safety regulation is clearly violated.

The next decision point (box 9-1.9) questions the presence of accident source terms. If an accident with reactor fuel damage occurs, radioactive material is released into the primary containment and transported into the suppression pool during the accident's blowdown phase. The RHR pumps take suction from the suppression pool during the initial phase of such an accident to supply makeup water to the reactor pressure vessel. The RHR piping will be contaminated by this radioactive material. If RHR is subsequently realigned into FPC mode, radioactive material may be transported to piping in the upper reactor building elevations and into the spent fuel pool. Radioactive material in such quantities in these areas has not been analyzed and represents the potential for exceeding off-site and control room radiation dose limitations. PP&L informed the NRC that SSES would not use RHR FPC on an accident unit with accident source terms present.[16] If Unit 1 spent fuel pool cooling cannot be restored using Unit 1's systems, the decision trees proceed to Figure 9-2 to determine if Unit 2 systems can be used.

Restoring SFP Cooling Using Unit 2 Systems

The decision tree on Figure 9-2 for cooling the Unit 1 spent fuel pool using Unit 2 systems is virtually identical to the decision tree on Figure 9-1 with three notable differences. First, the Unit 2 reactor building is more likely to be accessible than the Unit 1 reactor building following an accident on Unit 1. When an accident affecting Unit 1 is detected, the Unit 1 reactor building and the common refueling floor are automatically isolated from the Unit 2 reactor building. Therefore, the Unit 2 reactor building is probably accessible following an accident with fuel damage on Unit 1.

The second difference between Figures 9-1 and 9-2 involves the spent fuel pool configuration (box 9-2.2). The Unit 2 FPCS can be used to remove decay heat from the Unit 1 spent fuel pool if the fuel pools are cross-tied. The spent fuel pools are cross-tied when the gates between each pool and the cask storage pit are removed to form essentially one large combined pool as shown in Figure 9-8. Prior to the 10 CFR, Part 21 report, PP&L restricted the time that the fuel pools were cross-tied to a few weeks during refueling. PP&L imposed this limitation in response to an industry event in which a spent fuel pool was inadvertently drained. With the fuel pool gates installed, an inadvertent draindown affects only one spent fuel pool. PP&L recently committed to operate with the SSES spent fuel pools normally cross-tied. In order to normally operate in this configuration without increasing the risk of coupled spent fuel pool draindown, PP&L implemented measures to securely isolate the drain line at the bottom of the cask storage pit.[17]

The third difference between Figures 9-1 and 9-2 involves the heat removal capacity of the fuel pool cooling system (box 9-2.4). The capacity of the Unit 1 FPCS was not an issue in Figure 9-1 because it was implicitly assumed that the Unit 2 spent fuel pool was independently cooled. In Figure 9-2, the Unit 2 FPCS must handle the combined decay heat loads from both spent fuel pools. The fuel pool cooling systems are designed to handle the decay heat load from a single

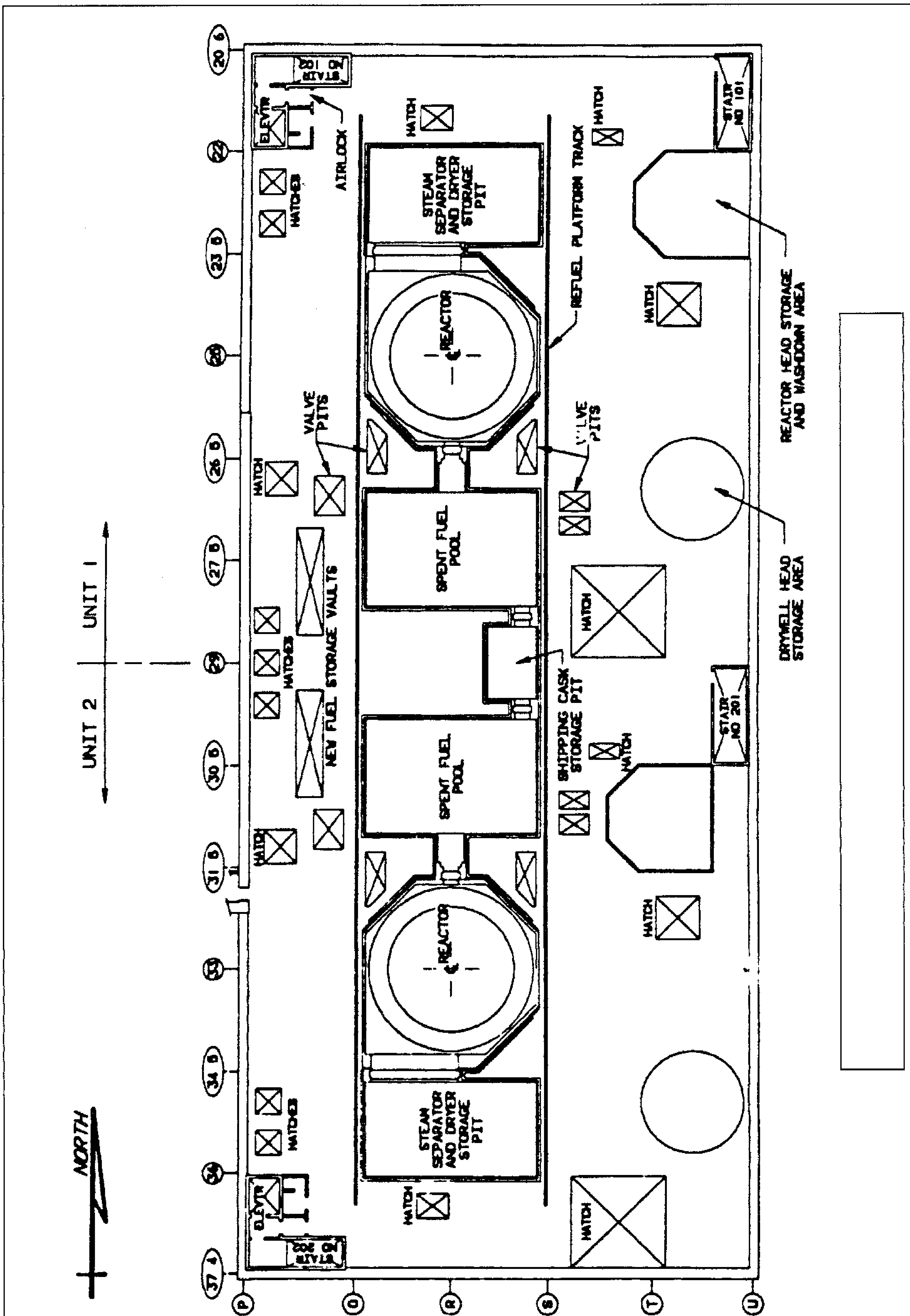

Figure 9-8 SSES Refueling Floor Plan

spent fuel pool filled to capacity. The likelihood that a single FPCS can handle the combined decay heat loads declines as the inventory in the spent fuel pools increases every refueling outage.

Box 9-2.7 questions access to the Unit 1 reactor building. In order for Unit 2 RHR FPC to cool the Unit 1 spent fuel pool, either the fuel pools must be cross-tied or the operator must have access to the Unit 1 reactor building to take manual actions to achieve the necessary system lineups.

Figures 9-1 and 9-2 indicate that the Unit 1 fuel pool boils if neither the Unit 1 nor the Unit 2 fuel pool cooling systems nor their backup systems can be used. The time required to reach boiling is primarily a function of the decay heat load in the spent fuel pool. The time to boil following a loss of fuel pool cooling is shown in Table 9-1. The actual decay heat loads in the SSES spent fuel pools as of 1994 were ≈$6x10^6$ BTU/hr on Unit 1 and ≈$5x10^6$ BTU/hr on Unit 2. Therefore, the spent fuel pool starts boiling 52 to 61 hours after the loss of cooling.

Table 9-1

Time (Hours) for Single Fuel Pool to Heatup to Specified Temperature Following Loss of Cooling from 110°F Initial Temperature

Decay Heat BTU/hr	125°F	140°F	180°F	Boiling	Boil-Off Rate, gpm
$3.0x10^6$	15.2	30.4	70.9	101.3	9.3
$5.0x10^6$	9.1	18.2	42.5	60.8	12.9
$7.0x10^6$	6.5	13.0	30.4	43.4	20.4
$10.0x10^6$	4.6	9.1	21.3	30.4	25.9
$12.6x10^6$	3.6	7.2	16.9	24.1	30.7
$13.2x10^6$	3.5	6.9	16.1	23.0	31.8

The design capacity of the SSES Fuel Pool Cooling System is $13.2x10^6$ BTU/hr—for any higher decay heat load, the FPCS would be unable to prevent spent fuel pool temperature from increasing and supplemental cooling would be necessary. The capacity of the SSES FPCS is reduced to $12.6x10^6$ BTU/hr to allow for plugging tubes in the fuel pool heat exchangers.

The 110°F initial fuel pool temperature is an administrative limit imposed at SSES and other nuclear power plants. If the spent fuel pool temperature exceeds 110°F, the refueling floor humidity and temperature produce an uncomfortable environment for workers and the condensation on equipment produces additional maintenance requirements. If the spent fuel pool temperature is initially less than 110°F, then the time to boil will be longer than the results shown in Table 9-1.

The SSES spent fuel pool has a surface area of approximately 1,350 ft^2.

Handling SFP Boiling Using Unit 1 Systems

Once the Unit 1 spent fuel pool boils, Figure 9-3 outlines the actions available to cope with this condition using Unit 1 systems. As with the actions to restore cooling, when the Unit 1 reactor building is not accessible (box 9-3.1), Unit 1's systems cannot be used to mitigate the boiling condition and efforts must be directed to Unit 2's systems.

Box 9-3.2 questions Unit 1's FPCS piping integrity. The consideration here is not the same as the concern on Figures 9-1 and 9-2 because FPCS piping integrity is not required in order for mitigating actions to be implemented. However, lack of FPCS piping integrity causes those actions to generate their own problems. Lack of FPCS piping integrity floods the Unit 1 reactor building with any water added to the spent fuel pool that overflows into the skimmer surge tank and spills out through the broken pipe. This flooding problem is eliminated if the pipe break is isolated by closing the manual valve (valve 153001 in Figure 9-7) between the seismic and non-seismic piping in the system.

Unit 1's ESW system can be used to provide makeup water to the boiling spent fuel pool when the Unit 1 reactor building is accessible. This makeup water replaces the water lost via boil-off. PP&L procedures call for the ESW makeup valves (the Unit 1 equivalent of valves 253500, 253091A, and 253090A or valves 253501, 253091B, and 253090B in Figure 9-7) to be opened (box 9-3.3), left open until the desired spent fuel pool level is attained (box 9-3.3a), and then closed (box 9-3.3b). This process is repeated when necessary. As with the time required to reach boiling, the boil-off rate shown on Table 9-1 is primarily a function of the decay heat load in the spent fuel pool.

The ESW makeup frequency to a boiling spent fuel pool is dictated by the range of the level instrumentation installed by PP&L after the 10 CFR, Part 21 report. This level instrument is mounted to the pool wall such that its indicating range extends a few inches above the normal pool surface and down several inches below the normal pool surface. The instrument's total range is 28 inches. If the instrument extends eight inches above the normal pool surface (as would be required to enable level monitoring prior to initiation of RHR FPC), then the level can drop 20 inches before the spent fuel pool level indication in the control room is downscale. Given a 14 gpm boil-off rate corresponding to a one inch per hour level drop and a 60 gpm ESW makeup rate, ESW makeup will be required every 25 hours.

Box 9-3.4 questions whether the operators can shutdown the reactor building heating, ventilation, and air conditioning (RB-HVAC) system to isolate Zone III from Zone I. Zone III is the refueling floor area while Zone I is the Unit 1 reactor building. The RB-HVAC system configuration determines the extent that structures and systems are exposed to the water vapor from the boiling spent fuel pool. The vapor will be confined to the refueling floor area when Zone III is isolated from Zone I, otherwise it will be transported throughout the Unit 1 reactor building.

When Zone III is isolated from Zone I, any airborne radiation in Zone III will be released unfiltered to the atmosphere. SSES is analyzed for a spent fuel pool boiling event with unfiltered airborne releases with results that are well within the 10 CFR, Part 100 limits. However, this analysis does not apply when there is airborne radiation in Zone III originating from sources other than the irradiated fuel assemblies in the spent fuel pools. There are numerous leakage pathways for airborne radioactivity from primary containment directly into Zone III. Therefore, Zone III cannot be isolated from Zone I following a DBA LOCA with fuel damage without incurring serious off-site dose consequences.

The Unit 1 reactor building will not experience the water vapor produced by the boiling spent fuel pool if Zone III is isolated from Zone I, but there will still be adverse consequences. Zone I has a free volume of $\approx 1.5\text{x}10^6$ ft^3 while Zone III has a free volume of nearly $2.7\text{x}10^6$ ft^3.[18] PP&L determined ambient temperature, pressure, humidity, and radiation exposure conditions for equipment in the reactor building assuming that Zones I and Zone III were connected and that the atmosphere within this combined volume was a homogeneous mixture. When Zone I is isolated from Zone III, the effective volume is reduced by more than half while the temperature, pressure, humidity, and radiation conditions are reduced negligibly. Isolating Zone III from Zone I unavoidably increases the temperature and radiation exposure levels within Zone I. As a result, components within the Unit 1 reactor building may fail.

The standby gas treatment system (SGTS) fails when Zone III is not isolated from Zone I during a boiling spent fuel pool event. The SGTS draws air from the secondary containment volume, passes it through filter trains with a 99% efficiency rate in retaining radioactive particulates, and discharges it into the atmosphere. The SGTS maintains the secondary containment at a slightly lower pressure than the atmosphere so that the only release pathway will be through its filter trains.

The SGTS inlet ductwork at SSES had fire dampers with fusible links that automatically closed when the process flow temperature exceeded 165°F. This isolation protected the charcoal in the filter trains from high temperature caused by a fire in the reactor building. The process flow temperature was calculated to approach 180°F in a postulated boiling spent fuel pool event.[19] Following the 10 CFR, Part 21 report, PP&L replaced the SGTS fusible links with ones rated at 285°F.

The 10 CFR, Part 21 report identified that the SGTS at SSES could fail even if the fire dampers remained open with the spent fuel pool boiling. Saturated air at 180°F drawn from the refueling floor air space by the SGTS would be transported through sheet metal ductwork in the reactor building where the external ambient temperatures are 85°F to 104°F. There would be significant condensation from the process flow before it reached the demisters and drains at the SGTS filter trains. The condensation accumulation could fail the SGTS by either collapsing the inlet ductwork from the weight of the collected water or blocking the flow path (i.e., a loop seal effect).

When the NRC staff looked into the SGTS operability concerns, it requested that PP&L evaluate the condensation issue. PP&L's analysis determined that the SGTS failed due to blockage of the flow path in as little as 17 hours after spent fuel pool boiling. This analysis prompted PP&L's commitment to normally operate with the SSES spent fuel pools cross-tied to reduce the spent fuel pool boiling probability.[20] PP&L recently installed low-point drains in the SGTS ductwork to reduce the challenge to SGTS operability following a boiling spent fuel pool event.

SGTS failure results in an immediate loss of secondary containment. The secondary containment failure is not cataclysmic—cracks will not appear in the concrete structure and doors will not blow off hinges, but it has immense repercussions. As soon as the SGTS fails, the off-site radiological consequences from an accident immediately increase by a factor of nearly 100 because the airborne effluent is no longer filtered before being released to the atmosphere. If the calculated population dose from an accident is merely 1% of the 10 CFR, Part 100 limit, that population dose approaches the limit after the SGTS fails. If the calculated population dose from an accident is only half of the 10 CFR, Part 100 limit, that population dose becomes nearly 50 times the limit after the SGTS fails.

The boiling spent fuel pool adversely affects far more than just the SGTS when Zone III is not isolated from Zone I. During most design basis events, the normal RB-HVAC system is isolated and it recirculates air throughout secondary containment to promote uniform mixing. In this condition, the RB-HVAC system does not draw in any outside air and it provides neither heating nor cooling. The large motors located in the reactor building that must operate are provided with room coolers. However, these room coolers are not designed for the latent heat and vapor released from a boiling spent fuel pool. The total heat load in the reactor building from all sources, except boiling spent fuel pool latent heat, is $\approx 5 \times 10^6$ BTU/hr following a DBA LOCA. This heat load increases by a factor of two or three if the latent heat from the boiling spent fuel pool is considered. Due to the increased temperature and moisture content in the reactor building, the room coolers will not transfer as much heat out of these vital areas and the motors may fail due to over-temperature effects.[21] One or more motor failures adversely affects both cooling irradiated fuel in the reactor core and primary containment cooling.

Another consequence from the boiling spent fuel pool on the reactor building when Zone III is not isolated from Zone I is flooding. PP&L calculated that 2.5 million gallons of water must be added to a boiling spent fuel pool over a 30 day period to compensate for the boil-off. The UHS contains sufficient volume and the ESW system can deliver this much water to a boiling spent fuel pool, but the reactor building is not designed to cope with the 2.5 million gallons of water from boil-off. Because the SGTS fails when the spent fuel pool boils, the vapor is not forcibly removed from the reactor building and eventually condenses on the metal siding that forms the walls of the refueling floor, on the reactor building's concrete walls, and on just about everything in the reactor building. Much of the condensation will travel through floor and equipment drains into the reactor building sumps.

The reactor building sumps are large, but they are not designed to hold 2.5 million gallons. The sumps will overflow and flood the basements in both reactor buildings unless the sump pumps are able to transfer the water outside secondary containment to the radwaste system. Following an accident with fuel damage, the sump pumps are disabled to prevent the spread of radioactivity outside secondary containment. The ECCS pumps located in the reactor building basement may be flooded when the sumps overflow. Equipment that does not fail from being submerged may fail due to moisture intrusion into cable connections, control switches, relays, breakers, and other electrical and mechanical components.

The irradiated fuel in the boiling spent fuel pool can be kept covered with water using the Unit 1 ESW system when the Unit 1 reactor building is accessible, but the potential consequences include loss of primary and secondary containment, failure of ECCS pumps on both units, and meltdown of the irradiated fuel assemblies in the reactor core.

Handling SFP Boiling Using Unit 2 Systems

Once the Unit 1 spent fuel pool boils, Figure 9-4 outlines the actions available to handle this condition using Unit 2 systems. As with the decision tree on Figure 9-2, accessibility to the Unit 2 reactor building is not an issue. The major difference between the decision tree on Figure 9-4 and the decision tree on Figure 9-3 is that either the spent fuel pools must be cross-tied (box 9-4.1) or the Unit 1 FPCS piping must be intact (box 9.4-2) in order for the Unit 2 ESW system to provide makeup water to a boiling Unit 1 spent fuel pool. When the spent fuel pools are cross-tied, then any ESW makeup to the Unit 2 pool maintains the level in the Unit 1 pool. When the spent fuel pools are isolated, the Unit 2 ESW system can fill the Unit 2 pool until it overflows and fills the Unit 2 skimmer surge tank. Continued Unit 2 makeup with the Unit 2 skimmer surge tank filled will cause backflow from the skimmer surge tank into the cask storage pit. Once the cask storage pit is filled, the overflow will be carried into the Unit 1 skimmer surge tank. Eventually, with both skimmer surge tanks and the cask storage pit filled, the ESW makeup to the Unit 2 pool will cascade into the Unit 1 pool as indicated in Figure 9-7. However, if the Unit 1 FPCS piping is not intact, the overflow from the cask storage pit into the Unit 1 skimmer surge tank will drain out through the broken pipe and never make it into the Unit 1 fuel pool.

The irradiated fuel assemblies in the boiling spent fuel pool can be kept covered with water using the Unit 2 ESW system when the fuel pools are cross-tied or the Unit 1 FPCS piping is intact as shown in Figure 9-4, but the potential consequences include loss of primary and secondary containment, failure of ECCS pumps on both units, and meltdown of the irradiated fuel assemblies in the reactor core.

Boiling SFP's Ultimate Consequences

Figure 9-5 outlines the sequence of events that occur if the Unit 1 spent fuel pool boils without sufficient makeup water. The water level gradually decreases as the inventory boils-off. The decreasing level causes radiation levels in the reactor building to increase as the shielding over the spent fuel assemblies is reduced. PP&L calculated a 100,000 Rem/hr dose rate at the spent fuel pool railing when the water level drops to five inches above the top of the storage racks. A person standing at the spent fuel pool railing under those conditions receives a lethal radiation dose in 16 to 22 seconds.

According to PP&L, the "design of the Susquehanna Station is inadequate to properly cope with the radiological consequences of a credible loss of water from the spent fuel pool."[22] The decreased SSES spent fuel pool level has not been thoroughly analyzed, and PP&L does "not know the effects on the power plant or upon the public."[23] The effects may not be known, but it is known that these effects will neither be beneficial to the power plant personnel nor to the public.

As the boil-off continues, the irradiated fuel assemblies in the storage racks gradually uncover. An unfortunate Catch-22 situation develops. If the fuel channels have been removed from the irradiated fuel assemblies, then the decay heat generated by spent fuel bundles discharged from the reactor more than four years ago may be removed by convection through air circulation. However, the recently discharged spent fuel bundles (with or without fuel channels) and perhaps some of the older spent fuel bundles with fuel channels still installed will not be sufficiently cooled. During heatup, the fuel assembly's cladding catches fire. The fire can propagate to the older bundles, particularly when the fuel channels have been removed.

In some respects, the slow boil-off of the fuel pool water inventory is worse than the sudden, complete drainage by the catastrophic failure of the spent fuel pool structure. When the spent fuel pool is completely drained, air is able to circulate under the storage racks and up through the irradiated fuel assemblies for optimum convective heat transfer. During the slow uncovering from boil-off, the air circulation past the exposed irradiated fuel assemblies, particularly those assemblies with fuel channels, is restricted and the heat transfer is limited to that provided by the steam flowing upward.

The radioactivity released from irradiated fuel assemblies damaged in the spent fuel pool proceeds unfiltered to the atmosphere. At best, the refueling floor and reactor building afford some holdup and retention. At worst, the radioactivity is released directly to the atmosphere.

The uncontrolled and unfiltered radioactivity release from a postulated boiling spent fuel pool event should be cause for concern, especially when one considers that each SSES spent fuel pool is licensed to store 2,840 irradiated fuel assemblies. Each SSES reactor core contains 764 fuel assemblies. The boiling spent fuel pool event at SSES therefore involves potential damage to nearly four times as much irradiated fuel as could possibly be damaged in the most severe reactor core accident. To make matters worse, this fuel damage occurs outside primary containment after secondary containment was either lost or bypassed.

The design deficiencies related to spent fuel pool cooling at the Susquehanna Steam Electric Station affect many, but not all, other operating boiling water reactors in the United States. The newest BWRs (Grand Gulf Nuclear Station, Clinton Power Station, and Perry Nuclear Power Plant) were designed and built with safety-related fuel pool cooling systems. Thus, the spent fuel assemblies in the spent fuel pools at these facilities can be cooled at all times, including following postulated design basis events. The Susquehanna concerns do not affect PWR plants because their spent fuel pools are located outside containment.

Events Leading to Discovery

In April 1990, PP&L initiated a power uprate project to obtain the licensing changes required to operate the SSES units at a higher power level. The NRC originally licensed each SSES unit for a maximum power level of 3,293 MWt. The units had been designed for a power level of 3,457 MWt to provide a conservative margin to the operating power level. The power uprate project's goal was to operate each SSES unit at the design power level, thus increasing generating capacity by approximately 50 MWe per unit.[24]

PP&L formed a power uprate project team of utility engineers and consultants. This team evaluated the nonnuclear steam supply systems (NSSS) and balance of plant (BOP) systems at the power update conditions. In some cases, the team developed calculations to support these evaluations.

The original calculation of SSES secondary containment heat loads following a postulated design basis loss-of-coolant accident (DBA LOCA) identified the heat sources (e.g., piping transmission, lighting, operating electrical equipment, etc.) and the heat sinks (e.g., room cooling units, etc.) in each room and area. This information was then used in steady-state heat transfer equations to determine the total secondary containment heat load and the individual room and area ambient temperatures.

Power uprate would cause the secondary containment heat loads to increase due to higher system process temperatures. An attempt to take existing secondary containment heat load calculation results and apply a parametric study for the power uprate conditions proved unsuccessful due to several incremental changes in room and area heat loads since the original calculation and minor errors and omissions noted in the original calculation. Therefore, two new calculations of secondary containment heat loads were commissioned—one for the existing SSES configuration and the second for the SSES configuration at the power uprate conditions.

During the review of the draft calculations in March 1992, it was noticed that the decay heat from the irradiated fuel assemblies in the spent fuel pools located within secondary containment was not considered. The draft calculations neither provided for the piping transmission heat loads from the fuel pool cooling system when the spent fuel pool was cooled nor provided for the latent heat from the spent fuel pool if it boiled. The original calculation had also omitted these heat loads. Since either fuel pool cooling or spent fuel pool boiling occurs following the postulated DBA LOCA, the original and the draft calculations were nonconservative because they neglected both phenomena.

The draft calculations were revised to assume that the FPCS operated when off-site power was available and to assume that the backup fuel pool cooling system operated when off-site power was not available. Because the SSES Final Safety Analysis Report (FSAR) described both spent fuel pools boiling following a postulated seismic event, even these revised draft calculations were nonconservative because they neglected the latent heat from the boiling spent pools.

PP&L commissioned an independent engineering evaluation of the boiling spent fuel pool deficiencies. This study validated the deficiencies and identified numerous other potential problems. These additional issues included the possibility that the nonseismic fuel pool cooling system piping fails due to the hydrodynamic loadings caused by a DBA LOCA and that the RHR fuel pool cooling assist mode (the only backup to the fuel pool cooling system) is unavailable due to other dedicated RHR system functions following an accident.[25]

Federal regulations compel individuals and companies to report nuclear safety problems to the NRC. Nuclear power plant owners are required under 10 CFR, Parts 50.72 and 50.73 to promptly inform the NRC when emergency equipment actuations and failures occur and when potential degraded conditions are discovered. Companies and individuals are required under 10 CFR, Part 21 to inform the NRC of generic safety issues. The intent of these regulations is twofold: first, to provide assurance that deficiencies at a nuclear power plant are corrected; and second, to enable deficiencies discovered at one plant to be fixed at other plants having the same vulnerabilities. The NRC was harshly criticized after the Three Mile Island accident for having known about the occurrence of the same accident precursor event at the Davis-Besse plant, but not having alerted operating plants with similar designs (including TMI).

PP&L formed a task force to conduct a thorough engineering evaluation of the spent fuel pool cooling concerns. The task force recommended several plant modifications and procedure changes to resolve the concerns. The task force, assuming that every recommendation was fully implemented (when in fact, none of these actions had yet been completed), concluded that SSES could mitigate all loss of spent fuel pool cooling events.[26] The task force, however, failed to address the plant's ability to cope with loss of spent fuel pool events in its existing condition.

PP&L submitted a report dated November 17, 1992, to the NRC on the Susquehanna concerns. The report was submitted as a voluntary report under 10 CFR, Part 50.9, rather than as a required LER under 10 CFR, Parts 50.72 and 50.73. PP&L indicated that they might undertake some measures to enhance SSES operation as a result of these problems.[27]

On November 27, 1992, the engineers who had originally discovered the design deficiencies in the Susquehanna Steam Electric Station related to spent fuel pool cooling reported a substantial nuclear safety hazard to the NRC in accordance with the provisions of 10 CFR, Part 21. Their objectives in submitting this report to the NRC were to ensure that the design deficiencies at SSES were properly resolved and to have the Susquehanna concerns evaluated generically because many of the 35 operating BWRs in this country potentially contained similar vulnerabilities.

NRC In Action (is that two words or three?)

The NRC initiated a deliberate assessment of the Susquehanna 10 CFR, Part 21 report. After corresponding and teleconferencing with PP&L on the subject over several months, the NRC appeared ready to accept PP&L's contention that these concerns involved minimal safety significance.

On March 18, 1993, PP&L reassured the NRC during a public meeting that the Susquehanna design was fully adequate as-is. Ironically, one of the 10 CFR, Part 21 authors was contacted that same day by an engineer at the Seabrook plant. A modification to Seabrook's fuel pool cooling system was planned and the utility obtained a copy of the 10 CFR, Part 21 report from the NRC. The engineer at Seabrook had discovered that many pages were missing from the report. The original report with its 35 attachments totalled nearly 300 pages. One of the report's authors had duplexed many of its attachments to reduce the document's thickness, then took it to a commercial printing shop to have copies made. Due to a mixup, the document had not been duplexed when copied. Flawed copies of the 10 CFR, Part 21 report were submitted to the NRC Chairman, the NRC's Regional Administrator - Region I, and the NRC's Senior Resident Inspector stationed at SSES. The depth of the NRC's inquiry into this reported nuclear safety hazard is indicated by the fact that they had not noticed the numerous missing pages after nearly four months of "intense scrutiny." In late March 1993, the heretofore missing pages were mailed to the engineer at Seabrook and to the NRC.

The authors of the Part 21 requested an opportunity to present their concerns in person to the NRC staff. The NRC staff arranged a public meeting. The Part 21 report's authors invited Senator Joseph Lieberman, Representative Richard Lehman, and Representative Philip Sharp, then chairmen of the three Congressional committees with NRC oversight, to attend this public meeting.

The report's authors outlined their concerns with the SSES design for spent fuel pool cooling to the NRC on October 1, 1993. They introduced decision trees very similar to Figures 9-1 through 9-5.[28] Congressmen Lieberman, Lehman, and Sharp did not attend this presentation, but their staffs had contacted the NRC staff prior to the meeting regarding the matter.

As a result of either the effectiveness of the presentation or the Congressional interest, the NRC issued Information Notice No. 93-83 to all operating BWRs on October 7, 1993.[29] This notification warned utilities of the concerns identified in the Susquehanna design, but did not require any actions to be taken. Nonetheless, this notice marked the first formal communication by the NRC staff to these operating BWRs regarding the generic concerns they had been evaluating for almost a year.

A week later, the NRC appointed a full-time project manager responsible for resolving the Susquehanna concerns. The following week, the NRC dispatched members from the Containment Systems and Severe Accident Branches to the Susquehanna site and the PP&L offices in Allentown, Pennsylvania, to investigate. In the final week of October 1993, the NRC's project manager and a senior NRC official went to Capitol Hill to brief Rep. Lehman's committee on their plans. Part of the NRC's prescribed actions was to develop a task action plan covering a systematic evaluation of the technical issues.

On March 14, 1994, the NRC invited the Part 21 report's authors to Rockville, Maryland, for a public meeting in which they presented their preliminary assessment of the Susquehanna concerns. The NRC staff acknowledged that the Susquehanna design failed to meet several federal safety regulations, but concluded that because PP&L had not explicitly discussed these deficiencies in the SSES FSAR and because the NRC had not detected these deficiencies at the time that the plant was originally licensed, these federal safety regulations no longer applied to Susquehanna.[30] This patently absurd decision has been called the "two wrongs make a right ruling." The NRC staff further explained that to now require PP&L to satisfy these federal safety regulations, that had been in effect years before SSES was licensed, constituted an imposition of new regulations that could only be done if proven cost-effective. Thus, the NRC put themselves in the untenable position of having to justify why federal regulations should be followed rather than requiring PP&L to justify why federal regulations should not be followed.[31]

On April 6, 1994, the authors of the 10 CFR, Part 21 report mailed a white paper to the governors, attorney generals, and United States Senators of 10 states with operating BWRs. This white paper outlined the design deficiencies that had been identified in the SSES design, expressed utter frustration at the direction the NRC staff was taking, and solicited their assistance in assuring the continued health and safety of the citizens in their states.[32] These state officials had no direct responsibility for nuclear power plant safety and could not have been blamed for taking no action, yet quite a few contacted the NRC about the issue.

Shortly after the white paper was distributed, the Susquehanna concerns came to the attention of the NRC's Advisory Committee on Reactor Safeguards (ACRS). The ACRS requested that the NRC brief them on the subject in May 1994. In their presentation before the ACRS, the NRC abandoned the "two wrongs make a right ruling" in favor of a probabilistic risk assessment (PRA) argument. The NRC claimed that the chances of the accident sequences were so small that their devastating consequences could be neglected. When the ACRS members asked pointed questions about the details of the NRC's PRA study, the NRC could not answer them. The ACRS requested that the NRC return when they were prepared to explain the basis behind their PRA study.[33]

Two weeks after the ACRS session, the NRC notified PP&L that they had concluded that SSES was operating outside its design basis because the SGTS would fail if the spent fuel pool boiled following a seismic event. PP&L had determined that at least one SSES spent fuel pool would boil following a seismic event that incapacitated the fuel pool cooling systems on both units. On June 1, 1994, PP&L committed to cross-tie the SSES spent fuel pools during normal operation to double the number of systems available to provide fuel pool cooling. PP&L also committed to implement administrative controls that keep the spent fuel pool time-to-boil greater than 25 hours following a postulated loss of cooling event to provide time for restoring fuel pool cooling. These commitments significantly reduce (but not eliminate) the probability of a boiling spent fuel pool at SSES.[34]

The Part 21 report's authors met with NRC Chairman Ivan Selin, at his invitation, on July 8, 1994. Dr. Selin listened to the Susquehanna concerns and questioned the NRC senior management officials in attendance about their generic implications. Dr. Selin's technical assistant said after this meeting, "I think there was a turning of the tide and it's recognized that they do have some valid points that need to be addressed."[35] The NRC had acknowledged that the Susquehanna concerns contained valid points when they issued Information Notice No. 93-83 to all operating BWRs in October 1993. Therefore, the turning of the tide must have been the NRC's recognizing that these valid points needed to be addressed.

The NRC issued its draft safety evaluation on the Susquehanna concerns on October 25, 1994. The NRC resurrected the "two wrongs make a right ruling" in determining that the SSES design deficiencies were outside the plant's licensing basis. The NRC additionally performed two PRA studies to determine if it would be cost effective to ask PP&L to correct the deficiencies. One PRA study purported to examine the SSES configuration at the time of the 10 CFR, Part 21 report, while the second PRA study examined SSES after the numerous plant modifications, procedure changes, training upgrades, and calculations had been completed. The NRC concluded that SSES was safe enough at the time of the 10 CFR, Part 21 report, and was much safer after all these changes had been made.[36]

The authors of the 10 CFR, Part 21 report provided the NRC with over 60 comments on serious errors and deficiencies in the draft safety evaluation on the Susquehanna concerns.[37] Many of these comments identified invalid assumptions in the PRA study for the as-found case. For example, the NRC assumed that operators would initiate corrective actions following a loss of spent fuel pool cooling event when fuel pool temperature reached 125°F. The NRC justified this assumption on the SSES Technical Specifications that limit fuel pool temperature to ≤125°F since the operators are trained to follow Technical Specifications. At the time of the 10 CFR Part 21 report, SSES had no indication of fuel pool temperature available in the control room. The only fuel pool temperature indication existed on a local panel on the refueling floor. It is extremely poor engineering judgment to assume that an operator will be stationed in front of a local panel, especially a local panel located in an potentially high radiation area, monitoring a nonsafety related parameter following a design basis event. The significance of this invalid assumption is that the NRC's PRA study determined that sufficient time existed to restore fuel pool cooling or implement actions to mitigate the consequences from a boiling spent fuel pool. That time is simply not available if the operator does not detect the problem as early as the NRC assumed. The NRC considered all of these comments to have immaterial effect on the PRA studies' results.[38]

The NRC unveiled its Task Action Plan on Spent Fuel Pool Storage Safety on October 26, 1994.[39] This Task Action Plan included NRC Team Inspections of five or six nuclear power plants during 1995 to ascertain whether the Susquehanna concerns applied to other operating sites. Based on the results from these inspections, the NRC will consider issuing generic communications or revising regulations if deemed necessary for public health and safety. These team inspections have been completed and the NRC is currently busy compiling the data collected and drafting a report.

But what about other operating BWR facilities affected by some or all of the Susquehanna concerns? Do these facilities need the hardware, procedure, and training changes that PP&L implemented at Susquehanna? Do these facilities have adequate temperature and level instrumentation to promptly detect a loss of spent fuel pool cooling under all design basis conditions? Do these facilities have sufficient administrative controls in place to maintain at least a minimum time-to-boil following loss of spent fuel pool cooling to allow sufficient time to implement restoration actions? What about single unit BWR sites that cannot cross-tie spent fuel pools to double their options for removing decay heat? What about BWR and PWR units that have shutdown and started down the decommissioning path, yet still have irradiated fuel assemblies in their spent fuel pools?

At the moment, these are rhetorical questions. The public interest will be best served if the NRC is able to properly answer these questions when the Task Action Plan on Spent Fuel Storage Safety is completed—and not answered with blatantly ludicrous "two wrongs make a right" rulings. These essay questions are difficult to answer, but post-mortem inquiries into a major accident will ask these questions and force proper responses. If nuclear safety was truly the NRC's paramount objective, these questions would have been completely answered (or at least asked) by now. It is sincerely hoped that a major accident is not needed to obtain these answers. Unfortunately, recent NRC actions (or inactions) indicate that it will either take a major accident or pressure from the Congress or the American public to spur them into doing the right thing.

1. David A. Lochbaum and Donald C. Prevatte to Thomas T. Martin, Regional Administrator, Region I, Nuclear Regulatory Commission, "Susquehanna Steam Electric Station, Docket No. 50-387, License No. NPF-14, 10CFR, Part 21, Report of Substantial Safety Hazard," November 27, 1992.

2. Nuclear Regulatory Commission, "Severe Accident Insights Report," NUREG/CR-5132, April 1988.

3. Nuclear Regulatory Commission Report NUREG-0800, Standard Review Plan, Chapter 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," Rev. 2, July 1981.

4. Pennsylvania Power & Light Company, Susquehanna Steam Electric Station Final Safety Analysis Report, Section 18.1.20.3.4.3, "Reactor Building Accessibility," Rev. 35, July 1984, p. 18.1-41.

5. Pennsylvania Power & Light Company Engineering Report NE-92-002 Rev. 0, "Loss of Fuel Pool Cooling Event Evaluation for EDR #G20020," October 29, 1992.

6. Office of Technology Assessment, "Comments on the Department of Energy's Mission Plan for the Civilian Radioactive Waste Management Program," Staff Paper Submitted to the House Committee on Energy and Commerce, Subcommittee on Energy Conservation and Power, August 30, 1985, p. 9.

7. J. V. Parrish, Assistant Managing Director, Operations, Washington Public Power Supply System, to United States Nuclear Regulatory Commission, "Nuclear Plant WNP-2, Operating License NPF-21, Licensee Event Report No. 93-018," May 28, 1993.

8. J. R. Miltenberger, Manager, Nuclear Safety Assurance Group, Pennsylvania Power & Light Company, to G. T. Jones, Manager - Nuclear Plant Engineering, Pennsylvania Power & Light Company, "Spent Fuel Pool Cooling," PLI-72367, September 9, 1992.

9. 10 CFR, Part 50, Appendix A, General Design Criterion 63, "Monitoring fuel and waste storage."

10. R. G. Byram, Senior Vice President, Nuclear, Pennsylvania Power & Light Company, to C. L. Miller, Project Director, Nuclear Regulatory Commission, "Susquehanna Steam Electric Station / Request for Additional Information on the Effects of a Loss of Spent Fuel Pool Cooling Event Following a Loss of Coolant Accident," PLA-3978, May 24, 1993.

11. The FPCS pumps can be supplied from the emergency power system at SSES, but the SWS pumps cannot. Since the SWS provides cooling water to the FPCS heat exchangers, power for the pumps in both systems is required for the FPCS to provide fuel pool cooling.

12. 10 CFR, Part 50, Appendix A, General Design Criterion 4, "Environmental and dynamic effects design basis."

13. R. G. Byram, Senior Vice President - Nuclear, Pennsylvania Power & Light Company, to C. L. Miller, Project Director, Nuclear Regulatory Commission, "Susquehanna Steam Electric Station / Information on Spent Fuel Pool Cooling Valve Testing Status," PLA-4186, August 8, 1994.

14. R. G. Byram, Senior Vice President, Nuclear, Pennsylvania Power & Light Company, to Nuclear Regulatory Commission, "Susquehanna Steam Electric Station / Loss of Spent Fuel Pool Cooling from Seismic Event / Use of RHR Fuel Pool Cooling Mode," PLA-4230, December 28, 1994.

15. Pennsylvania Power & Light Company Engineering Report NE-92-002 Rev. 0, "Loss of Fuel Pool Cooling Event Evaluation for EDR #G20020," October 29, 1992.

16. Transcript of Nuclear Regulatory Commission Public Meeting, "Meeting of NRC and Pennsylvania Power & Light on Fuel Pool Cooling," July 8, 1993.

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Appendix A

Spent Fuel Incidents

The following incident summaries supplement the discussion of spent fuel risks presented in Chapter 8. These summaries provide a representative sampling of the types and potential consequences of incidents that have occurred in the nuclear power industry.

Seal Failures

September 1972: An inflated seal at the transfer gate on the Point Beach Unit 1 spent fuel pool deflated when its air supply failed. Nearly 1,200 gallons leaked from the spent fuel pool, although no irradiated fuel assemblies were in the pool at the time.

October 1976: An inflated seal at the inner pool gate on the Brunswick Unit 2 spent fuel pool deflated due to an air leak and a power failure for the air compressor. The leakage caused the SFP level to drop five inches.

June 1980: The inflatable seal at the transfer gate on the Trojan spent fuel pool was not properly inflated prior to draining the refueling cavity. The leakage caused the SFP level to drop ten inches below the minimum level permitted by the plant's operating license.[1]

May 1981: After draining the fuel transfer canal with the transfer canal door closed and the door seal inflated on the Arkansas Nuclear One Unit 2 spent fuel pool, maintenance on the air system interrupted air supply to the seal. The leakage caused the SFP level to drop seven feet until the water levels between the spent fuel pool and the fuel transfer canal equalized. If the seal had leaked with the fuel transfer tube gate valve open and the fuel transfer tube blind flange removed, the spent fuel pool could have drained down to just above the top of the irradiated fuel assemblies.[2]

October 1984: The inflatable seal on the gate between the San Onofre Unit 2 spent fuel pool and the spent fuel shipping cask pit deflated following an air compressor failure. The backup air compressor failed to start. Nearly 20,000 gallons leaked from the spent fuel pool and dropped the SFP level 19-1/2 inches, although no irradiated fuel assemblies were in the pool at the time.[3]

December 1987: A valve in the return line to the refueling water storage tank at Wolf Creek was inadvertently left open, allowing the SFP level to drop to a minimum of 22 feet over the irradiated fuel assemblies during the next two days. The problem was not detected by the operators because the SFP level alarm was inoperable at the time.[4]

May 1988: The refueling cavity water seal at Surry Unit 1 failed with the reactor core fully offloaded into the spent fuel pool. The leakage dropped the SFP level about three feet.[5]

October 1988: The air supply line to the inflatable seal on the transfer canal door at Surry Unit 1 experienced a pinhole leak that was quickly detected and repaired. At the time, the fuel transfer canal was drained, the fuel transfer tube was open with the blind flange removed on the containment side and the gate valve open on the spent fuel pool side for testing in preparation for an upcoming refueling outage. Virginia Power determined that the spent fuel pool could have been drained to within 13 inches of the top of irradiated fuel assemblies in the storage racks. The radiation field on the refueling floor was estimated at 50 Rem/hr in that condition. If this postulated event occurred shortly after refueling instead of just prior to refueling, the radiation field would have been significantly higher.[6]

Loss of Fuel Pool Cooling

January 1977: A power interruption lasting two hours shut down the cooling pump for the spent fuel pool at General Electric's Midwest Fuel Recovery Plant. The outside temperature was -19°F at the time. When the power was restored, it was quickly discovered that a pipe in the cooling system had frozen and ruptured. The cooling system remained shut down for several weeks while the piping was repaired. Decay heat from the irradiated fuel assemblies heated the pool to an equilibrium temperature of 115°F. The humidity in the building rose to an uncomfortable level, but otherwise this incident had no adverse impact on either plant personnel or the general public.[7]

May 1992: A spent fuel pool cooling system failure at Comanche Peak Unit 1 went undetected for 17 hours until the NRC Resident Inspector noted discrepancies between log entries and control room indicators. Both Unit 1 spent fuel pool cooling trains were discovered to be inoperable. Texas Utilities Electric Company had to use cooling water from the unfinished and untested Unit 2 to cool the Unit 1 spent fuel pool.[8]

June 1993: Cooling water to the spent fuel pool heat exchangers at South Texas Unit 2 was inadvertently lost. This event remained undetected for 13 hours during a refueling outage with the reactor core fully off-loaded into the spent fuel pool. During the transfer of electrical power buses for maintenance activities, the component cooling water (CCW) system momentarily experienced a spurious surge tank low level signal. The signal closed the CCW supply and return valves to the spent fuel pool heat exchangers. The fuel pool cooling system pumps continued to circulate water through the heat exchangers although no heat removal was occurring. The SFP heated up from 99°F to 118°F during the 13 hours.

October 1993: Cooling water to the spent fuel pool heat exchangers at Farley Unit 2 was mistakenly isolated for three hours during a refueling outage with the reactor core fully off-loaded into the spent fuel pool. During motor-operated valve testing, the outlet valve on the in-service CCW heat exchanger was closed instead of the outlet valve on the standby heat exchanger. The SFP heated up from 90°F to 130°F in three hours. The annunciation of high SFP temperature at 130°F alerted operators to the problem. Spent fuel pool temperature peaked at nearly 140°F before cooling flow could be reestablished.

February 1995: The fuel pool cooling system at Indian Point Unit 3 was out of service for 2 hours and 15 minutes after a crane outside the Indian Point 2 protected area caused the 138 kV off-site feeder line for Unit 3 to arc. The emergency diesel generators started and supplied power to the safety related equipment at the plant. Following off-site power restoration, the nonsafety related fuel pool cooling system was returned to service.

Radiation Overexposure

Before 1972: An irradiated fuel assembly being withdrawn from the reactor core was inadvertently raised to within four feet of the water's surface. As the fuel assembly reached the normal upper limit position, the operator released the hoist's *UP* button. The hoist continued to raise the fuel assembly until the operator pressed the *STOP* button to de-energize the hoist motor. The radiation level at the operator's location was estimated to be 125 mr/hr with the irradiated fuel assembly at its maximum height. Investigation disclosed that repetitive jogging of the hoist had welded the *UP* relay contacts closed.[9]

April 1978: Two radiation protection technicians at the Trojan Nuclear Power Plant received whole body radiation doses of 27.3 and 17.1 Rem while performing a survey adjacent to an exposed section of the spent fuel assembly transfer tube.[10]

March 1984: A diver repairing the fuel transfer upender at the Palisades Nuclear Generating Plant received an exposure of about 4.5 Rem to the right thigh during a three dive series. The diver kneeled in a radioactive sludge layer on the tilt pit floor.[11]

Fuel Handling

1971: An irradiated fuel assembly became disengaged from the fuel handling tool and dropped approximately 10 feet into an empty fuel storage position during fuel transfer operations at a PWR. Visual examination did not disclose any discernible damage to the fuel assembly. The fuel handling tool's top edge showed some slight galling that prevented the tool from properly latching the fuel assembly.[12]

January 1974: While an irradiated fuel assembly was being transferred from the reactor core to the spent fuel pool at the Pilgrim Nuclear Plant, the main refueling grapple lowered without the operator's knowledge. The fuel assembly struck the reactor pressure vessel's side but was not damaged. Investigation disclosed that the *return-to-normal* spring in the grapple's control circuit had broken. After the operator raised the grapple to the fully retracted position and released the up-down switch, the switch fell through the neutral position to the *down* position due to the broken spring, causing the grapple to lower.

January 1974: While transferring an irradiated fuel assembly from the spent fuel pool to the channel inspection facility at the Pilgrim Nuclear Plant, the fuel assembly became detached from the main grapple and fell approximately 20 feet to the bottom of the spent fuel pool. Examination of the fuel assembly revealed that its channel slipped down over the nose piece and that the impact crushed the nose piece and lower end of the channel. There were no indications of broken fuel rods, and the fuel pool liner did not suffer damage. The grapple hook may not have been completely latched under the fuel assembly's handle allowing the operator to lift the fuel assembly with only a friction grip on its handle.[13]

January 1977: A fuel assembly was inadvertently released from the grapple at Peach Bottom Unit 3 and fell across the core. The assembly's release was attributed to inadvertent operation of the grapple open switch when the refueling mast controls had to be rotated away from the operator in conjunction with a slack cable signal when the fuel assembly nose cone contacted the core as it was being lowered, thereby satisfying all the interlocks for the grapple to open.

May 1977: A fuel assembly and mast were inadvertently dropped at Oyster Creek while lowering the assembly into a storage rack. The fuel and mast movement were arrested by the cable brake drum, without further damage, when the operator released the grapple lower lever. The drop resulted from shearing six bolts that coupled the refueling mast speed reducer to the cable drum. Examination indicated that all but two bolts had previously failed.[14]

December 1979: A new fuel assembly was dropped at Pilgrim while it was being transferred to its storage location in the spent fuel pool. The assembly was being transported with the reactor building overhead crane when it struck the top edge of the high-density fuel racks and the latching device on the auxiliary hook failed to retain the fuel assembly lifting bail. The assembly fell, striking the lifting bails on four spent fuel assemblies, then coming to rest on the top of the storage racks. The four spent fuel assemblies were not damaged. Spent fuel pool water samples were analyzed with no discernible change in activity levels detected.[15]

March 1981: During refueling at Millstone Unit 1, a new fuel assembly was dropped onto the upper core grid. It had just been placed in a core location. In the process of removing the fuel grapple and telescoping mast from this fuel assembly, the latching mechanism did not fully retract. Upon retraction of the mast, the fuel assembly was removed from the core in an unsecured condition to a point just above the upper core grid where it became loose and fell onto the upper core grid, coming to rest in a diagonal position about 35° above the horizontal. The operator failed to detect the fuel assembly on the grapple because he did not perform an adequate rotational check of the telescoping mast before raising it.

June 1981: A spent fuel assembly at Donald C. Cook Unit 1 was damaged when its lower end struck a ledge outside of the reactor pressure vessel. During lifting of the fuel assembly from the core, the two air lines and the electric cable serving the gripper tangled and caught the gripper-tube-up position switch, closing it before the gripper tube and attached fuel assembly were in the full-up position inside the canister. Closing the switch cleared the interlock allowing lateral movement of the manipulator crane. The lower end of the fuel assembly protruded below the canister and it struck a ledge on the refueling cavity floor just outside the reactor pressure vessel area. Several fuel rods in the assembly were damaged with one fuel rod dislodging from the assembly and falling onto the reactor cavity floor.[16]

February 1986: An irradiated fuel assembly was inadvertently lifted from the Haddam Neck core when the upper core support structure was removed from the reactor vessel. The irradiated fuel assembly stuck to the structure because of a bent fuel assembly locating pin. As the structure was moved laterally, the assembly struck the core barrel and dropped two to four feet onto the core. The dropped assembly and the two irradiated fuel assemblies it impacted were damaged.[17]

June 1994: While lowering a fuel assembly into the Quad Cities Unit 1 reactor core, the lower end-fitting caught on either the edge of the control rod blade guide or the upper core grid plate. The operator did not notice that the lower end-fitting was caught and continued to lower the fuel assembly. A fuel handling verifier noticed the problem when the fuel assembly was leaning 45° to 65° from vertical and notified the operator. The fuel assembly was raised, repositioned, and inserted into the core. After inserting the fuel assembly into the core, the refueling grapple would not release from the fuel assembly's bail handle. The fuel assembly was returned to the spent fuel pool. Inspection determined that the refueling mast grapple would not release because the bail handle had been bent. No other damage to the fuel assembly or adjacent core components was identified.

May 1995: As a storage cask loaded with 40 spent fuel assemblies was lifted out of the Prairie Island spent fuel pool, an overload device actuated to stop the crane's movement. The storage cask's bottom remained three inches below the top of the spent fuel pool's walls. The loaded cask remained suspended over the spent fuel pool for nearly 16 hours until station personnel manually bypassed the overload interlock to permit the crane to resume lifting.

June 1995: The lower tie plate and 41 fuel rods separated from the upper tie plate and the remaining eight fuel rods while an irradiated fuel assembly was being relocated within the Oyster Creek spent fuel pool. The fuel assembly had operated in the Oyster Creek reactor during the 1970s before being discharged in April 1980. A similar event occurred at Oyster Creek in 1986. Following the earlier event, the fuel rods were removed from the broken fuel assembly and placed into a defective fuel canister.[18]

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