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EXHIBIT 3

RISKS AND ALTERNATIVE OPTIONS ASSOCIATED WITH SPENT FUEL STORAGE AT THE SHEARON HARRIS NUCLEAR POWER PLANT

A report

prepared for

Orange County
North Carolina

by

Gordon Thompson

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About the author

Gordon Thompson is the executive director of IRSS. He received an undergraduate education in science and mechanical engineering, in Australia. Subsequently, he studied at Oxford University and received from that institution a doctorate of philosophy in mathematics in 1973.

During his professional career, Dr Thompson has performed technical and policy analyses on a range of issues related to international security, energy supply, environmental protection, and the sustainable use of natural resources. Since 1977, a significant part of his work has consisted of technical analyses of safety and environmental issues related to nuclear facilities. These analyses have been sponsored by a variety of nongovernmental organizations and local, state and national governments, predominantly in north America and western Europe. Dr Thompson has provided expert testimony in legal and regulatory proceedings, and has served on committees advising US government agencies.

About IRSS

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Abstract

Orange County, North Carolina, commissioned this report because the licensee of the Shearon Harris nuclear plant has requested an amendment of its operating license. The amendment would permit the activation of two currently unused spent fuel pools at Harris.

This report examines the risks and alternative options associated with spent fuel storage at Harris. The report identifies a potential for severe accidents at the Harris pools. Such accidents could release to the atmosphere an amount of cesium-137 an order of magnitude larger than the release from the 1986 Chernobyl accident. A severe accident at the Harris PWR, with containment failure or bypass, can be expected to initiate a large release from the fuel pools.

Alternative, safer options for spent fuel management are available. These options include dry storage of spent fuel, which is a well-established practice.

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1. Introduction

Carolina Power & Light Company (CP&L) requested, in December 1998, an amendment of its operating license for the Shearon Harris nuclear plant. The amendment, if granted by the Nuclear Regulatory Commission (NRC), would permit the activation of two currently unused spent fuel pools at Harris. In January 1999, Orange County commissioned this report, which examines the risks and alternative options associated with spent fuel storage at Harris.

Structure of this report

This report has two major components. One component is a main report which is comparatively brief and is intended for a non-specialist audience. The second component is a set of five appendices. These appendices contain detailed, technical material and citations to technical literature. Unless otherwise indicated, discussion in the main report rests upon the more detailed discussion in the appendices.

What is spent fuel?

Figure 1 shows a fuel assembly of the type that is used in the Harris reactor.[1] The fuel rods are 12 feet long, and the assembly is 8.4 inches square. After a fuel assembly is discharged from a reactor, it is "spent" in the sense that it can no longer be used to generate power. However, at this point in its life the assembly is much more dangerous than when it entered the reactor. It emits heat and intense radiation, and contains a large inventory of radioactive material.

Remainder of this report

The remainder of this main report begins with descriptions of the Harris plant (Section 2) and CP&L's intentions regarding the fuel pools at Harris (Section 3). Then, categories of potential accident at Harris are identified (Section 4), followed by descriptions of potential design-basis (Section 5) and severe (Section 6) accidents at the Harris pools. The offsite consequences of potential pool and reactor accidents are addressed in Section 7. Alternative options for spent fuel management are presented (Section 8), followed by a discussion of regulatory processes (Section 9). Conclusions are presented in Section 10.

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[1] Figure 1 is adapted from: A V Nero, A Guidebook to Nuclear Reactors, University of California Press, 1979, page 79.

2. Present status of the Harris nuclear plant

The Harris plant features one pressurized-water reactor (PWR). The core of this reactor contains 157 fuel assemblies, with a center-center distance of about 8.5 inches. The Harris plant was to have four units but only the first unit was built. (A unit consists of a reactor, a turbine-generator and associated equipment.) A fuel handling building was built to serve all four units. This building contains four fuel pools (A, B, C, D), a cask loading pool and three fuel transfer canals, all interconnected but separable by gates.

These pools and transfer canals allow spent fuel to be moved around and stored while remaining under water. The water provides cooling and also shields personnel and equipment from the radiation emitted by the fuel. Shipping casks can carry spent fuel to or from Harris. Casks are loaded and unloaded while submerged in the cask loading pool.

Pools A and B

Pools A and B contain fuel racks, and are in regular use. CP&L says that fresh fuel, and spent fuel recently discharged from the Harris reactor, is stored in pool A. Fuel examination and repair are performed in an open space in pool B. At present, pools C and D are flooded but do not contain racks. The cooling and water cleanup systems for pools C and D were never completed.

Currently, pools A and B store spent fuel from the Harris reactor and from CP&L's Brunswick plant and Robinson plant. The Brunswick plant has two boiling-water reactors (BWRs) while the Robinson plant has one PWR. Shipment of spent fuel from Brunswick and Robinson to Harris is said by CP&L to be necessary to allow sufficient capacity in the pools at Brunswick and Robinson so that the entire core can be removed from the reactor.

Pools A and B now have a combined, potential capacity of 3,669 fuel assemblies. The center-center distance in the racks in pools A and B is 10.5 inches for PWR fuel and 6.25 inches for BWR fuel. This is a much more compact pool storage configuration than was used when nuclear plants first entered service. The United States has no national storage site or repository for spent fuel, so CP&L is currently obliged to store fuel at its plant sites. Compact storage in the existing pools is a comparatively cheap option for on-site storage.

3. Proposed activation of fuel pools C and D

CP&L seeks an amendment to its operating license so that it can activate pools C and D at Harris. By activating these pools, CP&L expects to have sufficient storage capacity at its three nuclear plants to accommodate all the spent fuel discharged by the four CP&L reactors (the Harris and Robinson PWRs and the two Brunswick BWRs) through the ends of their current operating licenses.

Capacity and configuration of pools C and D

CP&L plans to install racks in pool C in three campaigns (approximately in 2000, 2005 and 2014), to create a total capacity in this pool of 3,690 fuel assemblies. Thereafter, CP&L plans to install racks in pool D in two campaigns (approximately in 2016 and at a date to be determined), to create 1,025 spaces. Thus, the ultimate capacity of pools C and D will be 4,715 fuel assemblies. The center-center distance in the racks used in these pools will be 9.0 inches for PWR fuel and 6.25 inches for BWR fuel. In pool C, the space between the outermost racks and the pool wall will be 1-2 inches.

The PWR racks in pools C and D will have a smaller center-center distance than the racks in pools A and B (9.0 inches instead of 10.5 inches). This highly compact arrangement allows more PWR fuel to be placed in a given pool area but also has adverse implications for safety.

Cooling and electrical supply for pools C and D

The water in a spent fuel pool must be cooled and cleaned. Cooling is performed by circulating pool water through heat exchangers, where its heat is transferred to a secondary cooling system. At Harris, the secondary cooling system is the component cooling water (CCW) system. When the Harris plant was designed, the intention was that pools C and D would be cooled by the CCW system for Unit 2. Also, electricity would have been supplied to the circulating pumps at pools C and D from the electrical systems of Unit 2. However, Unit 2 was never built and its CCW and electrical systems do not exist.

CP&L's current plan is to cool pools C and D by completing their partially built cooling systems and connecting those systems to the Unit 1 CCW system. Electricity will be supplied to pools C and D from the electrical systems of Unit 1. The Unit 1 CCW system already provides cooling to pools A and B and serves other, important safety functions. For example, the Unit 1 CCW system provides cooling for the residual heat removal (RHR) system and reactor coolant pumps of the Unit 1 reactor.

Independent support systems for pools C and D

During CP&L's planning for the activation of pools C and D, the company considered the construction of an independent system to cool these pools. Within that option, CP&L considered the further possibility of providing dedicated emergency diesel generators to meet the electrical needs of pools C and D if normal electricity supply were unavailable. Construction of an independent cooling system for pools C and D, supported by dedicated emergency diesel generators, could provide the level of safety that was associated with the original design concept for Harris. However, CP&L has not proceeded with this option.

Capacity of the Unit 1 CCW system

In its present form, the Unit 1 CCW system cannot absorb the additional heat load that will ultimately arise from activation of pools C and D. Over the first few years of pool use, while the heat load is comparatively small, CP&L proposes to exploit the margin in the Unit 1 CCW system. Subsequently, CP&L intends to upgrade the Unit 1 CCW system so that it can accommodate the full heat load from pools C and D, and can also accommodate an anticipated power uprate for the Unit 1 reactor.

Safety implications

In order to exploit the margin in the existing CCW system so as to cool pools C and D, CP&L may be obliged to require its operators to divert some CCW flow from the RHR heat exchangers during the recirculation phase of a design-basis loss-of-coolant accident (LOCA) event at the Harris reactor. This is a safety issue because, during the recirculation phase of a LOCA, operation of the RHR system is essential to keeping the reactor core and containment in a safe condition. CP&L's exploitation of the margin in the existing CCW system is deemed by CP&L and NRC to constitute an "unreviewed safety question".

Lack of QA documentation

Activation of pools C and D will require the completion of their cooling and water cleanup systems, and the connection of their cooling systems to the Unit 1 CCW system. CP&L states that approximately 80 percent of the necessary piping was completed before the second Harris reactor was cancelled. However, some of the quality assurance (QA) documentation for the completed piping is no longer available. Much of the completed piping is embedded in concrete and is therefore difficult or impossible to inspect. To address this situation, CP&L proposes an "alternative plan" to demonstrate that the previously completed piping and other equipment is adequate for its purpose. Nevertheless, the cooling systems for pools C and D will not satisfy prevailing code requirements.

4. Types of potential accident at the Harris plant

Most of the radioactive material at the Harris plant is either in the reactor or in the spent fuel pools. Thus, these locations are of primary concern when one considers the potential for accidents. This report focusses on the potential for accidents in the reactor or the pools. At present, pools C and D at Harris pose no accident potential, because they are unused.

Some potential accidents could cause injury to plant personnel, without causing any offsite effects. Other potential accidents could release radioactive material beyond the plant boundary, causing offsite effects. The radioactive material could be released as an atmospheric plume, or into ground or surface waters. This report focusses on accidents that release an atmospheric plume which travels beyond the plant boundary. Such a plume will contain radioactive material in the form of gases and small particles. As the plume travels downwind, the small particles will be deposited onto land, bodies of water, structures and vegetation.

Design-basis and severe accidents

A nuclear plant is designed to accommodate the effects of a specified set of accidents, known as "design-basis" accidents. If the plant is properly designed and constructed, if its equipment and operators function in the required manner, and if external influences (e.g., earthquakes) do not exceed specified levels, then the offsite effects of a design-basis accident will be small. Design-basis accidents and their anticipated effects are described in a Final Safety Analysis Report (FSAR) prepared and regularly updated by the licensee.

In the early years of the nuclear industry, some people equated design-basis accidents with "credible" accidents. However, research and operating experience soon revealed that accidents more severe than the design basis are credible. The first systematic study of the potential for severe accidents was the Reactor Safety Study, completed and published by the NRC in 1975. "Severe" accidents are conventionally defined as accidents involving substantial damage to fuel, with or without a substantial release of radioactivity to the environment.

The Three Mile Island (TMI) reactor accident of 1979 was a demonstration of the potential for severe accidents. Soon thereafter, the NRC promulgated regulations which require an emergency response plan for each nuclear plant. These plans allow for large releases of radioactive material, of the kind that were identified in the Reactor Safety Study. The Chernobyl reactor accident of 1986 further demonstrated the potential for severe accidents. While the TMI accident released a small fraction of the reactor core's inventory of radioactivity, the release fraction during the Chernobyl accident was large.

Since the TMI accident, the NRC's safety regulation of nuclear plants has been guided by a hybrid set of assumptions. Many areas of safety regulation rely upon the assumption that accidents will remain within the design basis. Other areas, such as emergency response planning, assume that severe accidents can occur.

Pool-reactor interactions

At the Harris plant, the reactor and the fuel pools are adjacent, and they share support systems such as the Unit 1 CCW system and the emergency diesel generators. Thus, it is important to understand if an accident at the Harris reactor could accompany, initiate or exacerbate an accident at the Harris pools, or vice versa. The NRC has been slow to examine the potential for safety interactions between reactors and fuel pools. Neither CP&L nor the NRC has assessed the potential for these interactions at Harris.

PRAs and IPEs

A discipline known as probabilistic risk assessment (PRA) has been developed to examine the probabilities and consequences of potential accidents at nuclear facilities. PRA techniques are most highly developed in their application to reactor accidents, but can be applied to fuel pool accidents. Appendix B describes the characteristics, strengths and limitations of PRA.

CP&L has prepared a Level 2, internal-events PRA for the Harris reactor, in the form of an Individual Plant Examination (IPE). Also, CP&L has performed a limited assessment of the vulnerability of the Harris reactor to earthquakes and in-plant fires, in the form of an Individual Plant Examination for External Events (IPEEE).

The Harris IPE and IPEEE could be extended to encompass fuel pool accidents as well as reactor accidents. Such an extension would be logical, because there are various ways in which a severe accident or a design-basis accident at the Harris reactor might accompany, initiate or exacerbate an accident at the Harris fuel pools, or vice versa. However, there is no current indication that CP&L will extend the IPE or IPEEE, or will otherwise apply PRA techniques to potential accidents at the Harris fuel pools.

5. Design-basis pool accidents

The Harris FSAR considers two types of design-basis accident in the Harris fuel pools. One type of accident involves the dropping of a fuel assembly, while the other type involves the dropping of a shipping cask (but not into a fuel pool). In both cases, the FSAR estimates that the release of radioactivity would be relatively small. This report does not review the FSAR analysis.

In its license amendment application, CP&L has considered some other potential accidents, including the dropping of a rack or a fuel pool gate.[2] CP&L's analysis of these accident scenarios is limited in scope. Accidents of this type may be in an intermediate class of severity, and that potential class deserves further analysis.[3] This report focusses on the potential for severe accidents.

It should be noted that the use of pools C and D at Harris will involve many additional cask, fuel and rack movements. These additional movements will increase the cumulative probability of accidents associated with such movements.

6. Severe pool accidents

Spent fuel is stored in a compact, high-density configuration in pools A and B at Harris. CP&L's proposed activation of pools C and D will involve an even higher density of storage. Such high-density configurations inhibit heat loss from the fuel if water is partially or totally lost from a pool. As a result, partial or total loss of water can lead to an exothermic (heat-producing) reaction of the fuel cladding with air or steam. Such a reaction could liberate a large amount of radioactive material from the fuel.

Thus, two questions become important. First, what circumstances could cause a partial or total loss of water? This question is addressed in Appendix C. Second, will an exothermic reaction be initiated if water is lost? That question is addressed in Appendix D.

Potential for loss of water

A variety of events could cause partial or total loss of water from the Harris pools. These events deserve the level of analysis that would be provided by a thorough PRA. Performing a pool accident PRA is beyond the scope of our

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[2] License amendment application, Enclosure 7.
[3] A potential accident in this class, which deserves analysis, would involve the placement of a low-burnup or high-enrichment PWR assembly in the racks in pools C or D.

present work for Orange County. Here, the focus is on two types of event -- a reactor accident, and a sabotage/terrorism event. Consideration of these events demonstrates clearly that loss of water from the Harris pools is a credible accident.

The Harris IPE -- prepared by CP&L -- examines the potential for severe accidents at the Harris reactor. It identifies a category of severe accidents that would involve failure or bypass of the reactor containment. The IPE estimates the collective probability of accidents in this category to be 1 per 100,000 reactor-years.[4] Occurrence of accidents in this category would contaminate the plant with radioactivity, to the point where personnel access would almost certainly be precluded. Water would then be evaporated from the fuel pools, and fuel would be uncovered after a delay of perhaps 10 days.

A credible sabotage/terrorism event at Harris would involve a group taking control of the fuel handling building, shutting down the pool cooling systems, and siphoning water from the pools. The group would require military skills and equipment to take control of the fuel handling building. Siphoning water from the pools would be a comparatively easy task. Escape by the group would be difficult but not impossible. The probability of this event cannot be predicted by PRA techniques.

Initiation of exothermic reactions, given water loss

Since the late 1970s, the NRC has sponsored and performed a variety of studies that have examined the outcomes of a loss of water from a fuel pool. These studies have focussed almost entirely on the instantaneous, total loss of water from a pool. Computer models have been developed to investigate this situation. For a high-density pool configuration, current models suggest that an exothermic reaction will be initiated in fuel aged up to 1-2 years after discharge from a reactor. These models have not been applied to the specific configuration of the Harris pools.

Partial loss of water can be expected in many scenarios, rather than instantaneous, total loss of water. Partial loss of water can be a more severe situation, because convective heat transfer from fuel assemblies is inhibited. The NRC has neglected this issue. Preliminary analysis suggests that partial water loss could initiate an exothermic reaction in fuel aged 10 years after discharge.

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[4] This probability estimate should be accompanied by a range of uncertainty. Even with the inclusion of uncertainties, PRA-derived estimates represent lower bounds to actual accident probabilities.

An exothermic reaction could propagate from one set of fuel assemblies to an adjacent set of assemblies that might not otherwise suffer such a reaction. The NRC's studies of propagation are incomplete, but they acknowledge the potential for propagation.

Exothermic reactions in the Harris pools

CP&L representatives have stated that spent fuel assemblies will not be placed in pools C and D at Harris until the assemblies have aged for 5 years after discharge. However, there is nothing in CP&L's license amendment application that prohibits the placement of more recently-discharged fuel in pools C and D. In any case, preliminary analysis suggests that partial water loss could initiate an exothermic reaction in fuel aged 10 years after discharge. Thus, exothermic reactions could occur in pools C and D.

For the purpose of estimating the potential consequences of a pool accident at Harris, this report considers two scenarios for exothermic reactions. One scenario involves fuel aged up to 3 years after discharge from a reactor, while the second scenario involves fuel aged up to 9 years after discharge from a reactor. In both cases, it is assumed that the entire inventory of cesium in the affected fuel assemblies would be released to the atmosphere. This assumption is consistent with NRC studies.

7. Consequences of potential pool and reactor accidents

This report focusses on accidents that release an atmospheric plume which travels beyond the plant boundary. The consequences of such a release can be estimated by site-specific computer models. Here, a simpler approach is used, but this approach is adequate to show the nature and scale of expected consequences. The approach is described in Appendix E.

The role of cesium-137

The consequences of a pool accident can be adequately illustrated by examining a release of only one radioisotope – cesium-137. This isotope has a half-life of 30 years and is liberally released from damaged fuel. It dominates the offsite radiation exposure from the 1986 Chernobyl accident, and is a major contributor to radiation exposure attributable to fallout from the atmospheric testing of nuclear weapons in the 1950s and 1960s.

Three atmospheric releases of cesium-137 are postulated here for the purpose of examining consequences. First, a release of about 2 million Curies (2 MCi) corresponds to the most severe reactor accident identified in the Harris IPE. Second, a release of about 20 million Curies (20 MCi) corresponds to a pool accident affecting fuel aged up to 3 years after discharge from a reactor. Third, a release of about 70 million Curies (70 MCi) corresponds to a pool accident affecting fuel aged up to 9 years after discharge from a reactor.

Land contamination by cesium-137

Accident consequences are illustrated here by estimating the area of land that would be contaminated by cesium-137 to a level such that inhabitants would suffer an external radiation dose in excess of 10 rem over 30 years.[5] An exposure of 10 rem over 30 years would represent about a three-fold increase above the typical level of background radiation (which is about 0.1 rem/year). In its Reactor Safety Study, the NRC used a threshold of 10 rem over 30 years as an exposure level above which populations were assumed to be relocated from rural areas. The same study used a threshold of 25 rem over 30 years as a criterion for relocating people from urban areas, to reflect the assumed greater expense of relocating urban inhabitants.

In an actual case of land contamination in the United States, the steps taken to relocate populations and pursue other countermeasures (decontamination of surfaces, interdiction of food supplies, etc.) would reflect a variety of political, economic, cultural, legal and scientific influences. It is safe to say that few citizens would calmly accept a level of radiation exposure which substantially exceeds background levels.

For typical meteorology, a release of 2 MCi would contaminate 4,000-5,000 square kilometers of land, A release of 20 MCi would contaminate 50,000-60,000 square kilometers. Finally, a release of 70 MCI would contaminate about 150,000 square kilometers of land. Note that the total area of North Carolina is 136,000 square kilometers and the state's land area is 127,000 square kilometers.

Health effects of radiation

There is ongoing debate about the health effects of radiation at comparatively low doses. According to estimates by the National Research Council's BEIR V committee, a continuous exposure throughout life at a rate of 0.1 rem/year (above background) will increase the number of fatal cancers, above the normally expected level, by 2.5 percent for males and 3.4 percent for females, with an average of 16-18 years of life lost per excess death. If the dose-response function were linear, it would follow that continuous, lifetime exposure to 1 rem/year would increase the number of fatal cancers by 25

[5] Without countermeasures such as interdiction of food supplies, the internal dose could be of a similar magnitude to the external dose.

percent for males and 34 percent for females. The shape of the dose-response function is a subject of debate.

8. Alternative options for spent fuel management

The present mode of spent fuel storage in Harris pools A and B poses a major hazard. This hazard will be substantially increased if pools C and D are activated. CP&L has not properly characterized the present and potential hazard, nor has the company provided a systematic assessment of alternative options.

A situation like this calls for a systematic, comprehensive assessment of alternative options and their impacts. A full range of alternatives should be identified, and their impacts and other characteristics should be assessed. Performance of such an analysis is beyond the scope of the author's current work for Orange County. An abbreviated discussion is presented here.

Options not reviewed here

One option would be to cease operation of CP&L's nuclear plants. That option, which could be combined with other options for storage of CP&L's present stock of spent fuel, is not reviewed here. Another set of options would employ high-density pool storage but would introduce technical measures that sought to increase the reliability of the cooling systems for some or all of the Harris pools, or to decrease the potential for safety interactions between the pools and the reactor. Independent support systems for pools C and D, as mentioned in Section 3, would be in this class of options. Such options are not reviewed here.

Options reviewed here

This report focusses on two classes of options for spent fuel storage. One class involves dry storage of spent fuel, using proven technology. The second class, which could complement dry storage, involves low-density storage in pools. A combination of dry storage and low-density pool storage could offer a practical, proven means of dramatically decreasing the hazard posed by high-density pool storage at Harris.

Dry storage

The NRC has approved a variety of designs for the dry storage of spent fuel. These designs are described in Table 1, and their current use by licensees is described in Table 2.[6] It will be noted from Table 2 that a dry storage installation is licensed at CP&L's Robinson plant. This installation employs eight NUHOMS-7P modules, each of which can hold 7 fuel assemblies. All eight modules are fully loaded.[7]

Dry storage could be implemented at any of CP&L's three plant sites. This report does not recommend any particular design, but notes that the designs vary in their level of safety and other features. For example, some designs are more resistant to sabotage than others.

All of the approved dry storage designs are safe in the event that access to the plant site is precluded by the release of radioactive material during a reactor accident. None of the designs requires active cooling, electricity or operator attention. A sabotage/terrorism event at a dry storage installation could release only a small fraction of the radioactive material that could be released by a sabotage/terrorism event at the Harris pools in their present and proposed configuration. Overall, dry storage poses a much lower level of hazard than high-density pool storage, for the same quantity of fuel.

At present, the NRC licenses dry storage installations for only 20 years. However, the technology is capable of storing fuel for much longer periods. If CP&L employs the dry storage option, they should choose a design that has this capability. This choice, properly documented and supported by ongoing testing, would establish the basis for a license extension in the future.

Low-density pool storage

Spent fuel can be stored in pools in a low-density, open-rack configuration, as was common practice when nuclear plants were first operated. Given a sufficiently low-density configuration, partial or total uncovering of the fuel will not initiate an exothermic reaction in the fuel cladding, even for recently discharged fuel. The fuel would remain vulnerable to consolidation through a cask drop into a pool or a severe earthquake which disrupts the fuel racks. If such consolidation were accompanied by partial or total uncovering, an exothermic reaction could occur in the consolidated region. However, it is unlikely that this reaction would be propagated to other regions of a pool.

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[6] Tables 1 and 2 are adapted from: US Nuclear Regulatory Commission, Information Digest, 1998 Edition, NUREG-1350, Volume 10, November 1998.
[7] M G Raddatz and M D Waters, Information Handbook on Independent Spent Fuel Storage Installations, NUREG-1571, December 1996.

Summary

CP&L could employ a spent fuel storage strategy which combines dry storage with low-density pool storage. Some or all of pools A, B, C and D at Harris would be used in a low-density configuration. If appropriately designed and implemented, this strategy could dramatically reduce the hazard posed by present and proposed fuel storage arrangements at Harris.

9. Addressing risks and alternatives in the regulatory arena

Orange County has requested the NRC to hold a hearing regarding CP&L's license amendment application, and the NRC has established a Licensing Board for this case. These actions have initiated a regulatory process which has been employed many times before. A review of this process is beyond the scope of this report, but some brief observations may be helpful.

The licensing process will typically assume that regulatory decisions taken in the past were correct. Thus, the existing operations at Harris pools A and B might be held to establish a precedent for the proposed operations at pools C and D. However, this report shows that the NRC has not properly analyzed the potential for severe pool accidents at a generic level. This point may or may not influence the NRC's regulatory process, but it deserves continuing emphasis through all available channels.

At Harris, and nationwide, there is a need for a thorough assessment of the hazards associated with high-density pool storage, and of alternative options which could pose a lower hazard. Orange County would provide an important public service if it could persuade the NRC or another body to conduct such an assessment, perhaps in the form of an environmental impact statement. There has been discussion about the US Department of Energy taking title to the nation's spent fuel, while the fuel remains at plant sites. This move could provide an opportunity for a thorough assessment of risks and options, and for the adoption of safer means of fuel storage.

10. Conclusions

C1 Given the present and proposed configuration of spent fuel storage in the Harris pools, partial or total loss of water from the pools could initiate exothermic reactions of fuel cladding, in any or all of pools A, B, C and D.

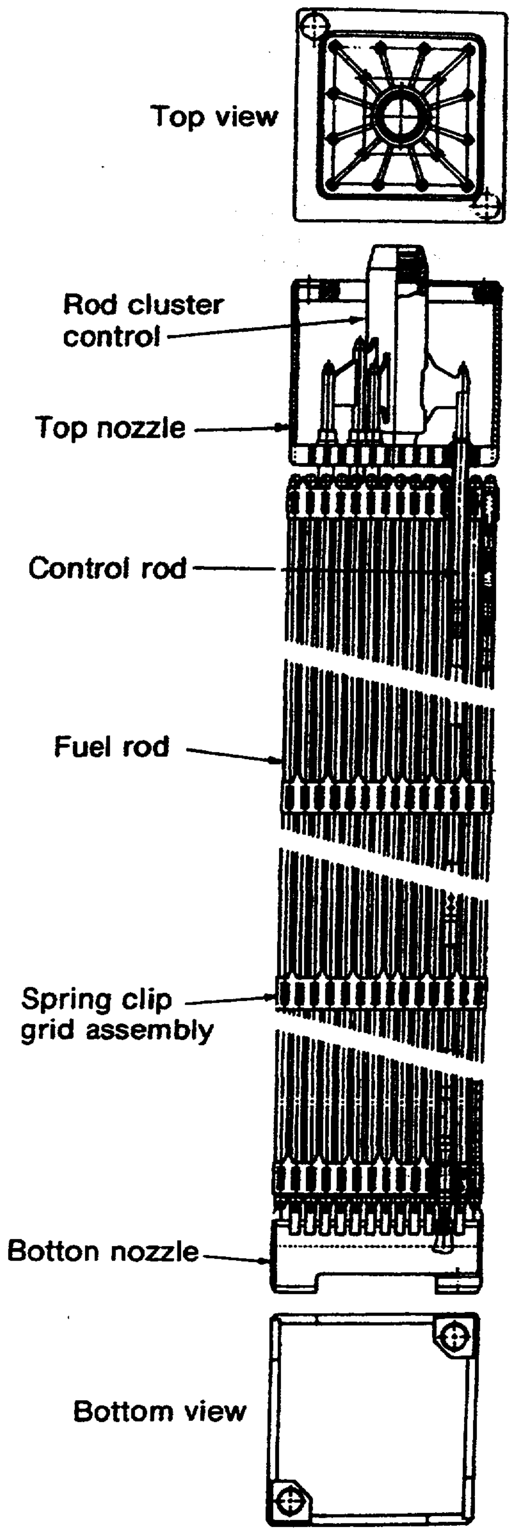
C2 Partial or total loss of water from the Harris pools could occur through a variety of events including acts of malice, and would be an almost certain outcome of a severe reactor accident at Harris involving containment failure or bypass; CP&L estimates the probability of the latter event as 1 per 100,000 reactor-years.

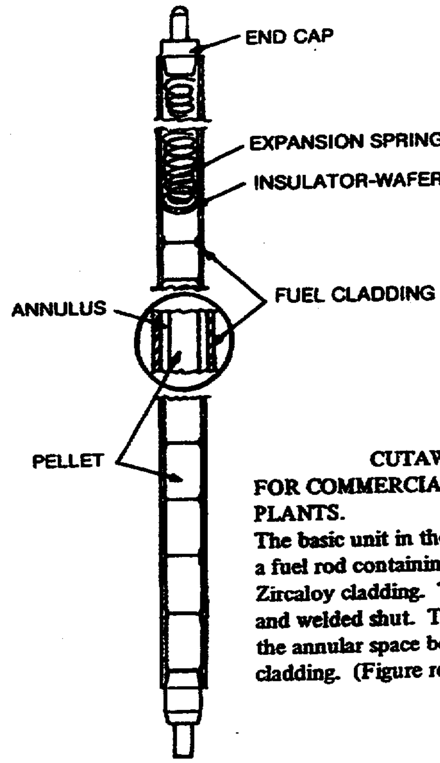
C3 Exothermic reactions in the Harris pools could release to the environment an amount of cesium-137 at least an order of magnitude larger than the amount released by the most severe potential accident at the Harris reactor.

C4 A large release of cesium-137, as could occur from exothermic reactions in the Harris pools, could significantly contaminate an area of land equal to the area of North Carolina.

C5 The probability and magnitude of a potential release from Harris of radioactive material in spent fuel could be dramatically reduced if CP&L adopted a fuel storage strategy which combines dry storage with low-density pool storage; this strategy would employ proven technology.

C6 Activation of pools C and D at Harris could increase the probability and magnitude of design-basis or severe accidents at the Harris fuel pools or reactor.





CUTAWAY VIEW OF OXIDE FUEL FOR COMMERCIAL LWR POWER PLANTS.
The basic unit in the core of a light-water reactor is a fuel rod containing uranium oxide pellets in a Zircaloy cladding. The rod is filled will helium gas and welded shut. The circled portion exaggerates the annular space between the pellet and the cladding. (Figure reproduced from WASH-1250.)

FUEL ASSEMBLY FOR A PRESSURIZED-WATER REACTOR.
In a pressurized-water reactor, fuel rods are assembled into a square array, held together by spring clip assemblies and by nozzles at the top and bottom. The structure is open, permitting flow of coolant both vertically and horizontally. All the assemblies in the reactor may have the same mechanical design, including prov.sion for passage of a control rod cluster (shown in the figure). Where there is no cluster, these positions may have neutron sources, burnable poison rods, or plugs. (Figure reproduced from WASH-1250.)

Figure 1

Fuel for a pressurized-water reactor

Vendor	Storage Design Model	Capacity (Assemblies)	Storage Design Approval Date	Certificate of Compliance Approval Date
General Nuclear Systems, Incorporated	Metal Cask			08/17/1990
	CASTOR V/21	21 PWR	09/30/1985	
	CASTOR X/28	28 PWR	04/22/1994	
	CASTOR X/33	33 PWR	11/24/1995	
Transnuclear, West Incorporated	Concrete Module NUHOMS-7P	7 PWR	03/28/1986	
Westinghouse Electric	Metal Cask MC-10	24 PWR	09/30/1987	08/17/1990
FW Energy Applications, Incorporated	Concrete Vault Modular Vault Dry Storage (MVDS)	83 PWR or 150 BWR	03/22/1988	
NAC International, Inc.	Metal Cask NAC S/T	26 PWR	03/29/1988	08/17/1990
NAC International, Inc.	Metal Cask NAC-C28 S/T	28 Canisters (fuel rods from 56 PWR assemblies)	09/29/1988	08/17/1990
Transnuclear, Incorporated	Metal Cask			11/04/1993
	TN-24	24 PWR	07/05/1989	
	TN-32	32 PWR	11/07/1996	
NAC International, Inc.	Metal Cask NAC-128/ST	28 PWR	02/01/1990	
Sierra Nuclear Corporation	Ventilated Cask VSC-24	24 PWR	03/29/1991	05/03/1993
Transnuclear West, Inc.	Concrete Module Standardized		04/21/1989	01/18/1995
	NUHOMS-24P	24 PWR		
	NUHOMS-52B	52 BWR		
NAC International, Inc.	NAC-STC	26 PWR	07/17/1995	

Note: PWR - Pressurized-Water Reactor; BWR - Boiling-Water Reactor

Table 1

NRC-approved dry spent fuel storage designs

Reactor Name Utility	Date Issued	Vendor	Storage Model
Surry 1, 2 Virginia Electric & Power Company	07/02/1986	Generals Nuclear Systems, Incorporated	Metal Cask CASTOR V/21 TN-32 NAC-128 CASTOR X/33 MC-10
H. B. Robinson 2 Carolina Power & Light Company	08/13/1986	Transnuclear West, Incorporated	Concrete Module NUHOMS-7P
Oconee 1, 2, 3 Duke Energy Company	01/29/1990	Transnuclear West, Incorporated	Concrete Module NUHOMS-24P
Fort St. Vrain* Public Service Company of Colorado	11/04/1991	FW Energy Applications, Incorporated	Modular Vault Dry Store
Calvert Cliffs 1, 2 Baltimore Gas & Electric Company	11/25/1992	Transnuclear West, Incorporated	Concrete Module NUHOMS-24P
Palisades Consumers Energy	Under General License	Pacific Sierra Nuclear Associates	Ventilated Cask VSC-24
Prairie Island 1, 2 Northern States Power Company	10/19/1993	Transnuclear West, Incorporated	Metal Cask TN-40
Point Beach Wisconsin Electric Power Company	Under General License	Sierra Nuclear Corporation	Ventilated Cask VSC-24
Davis-Besse Toledo Edison Company	Under General License	Transnuclear West, Incorporated	Concrete Module NUHOMS-24P
Arkansas Nuclear One Entergy Operations	Under General License	Sierra Nuclear Corporation	Ventilated Cask VSC-24
North Anna Virginia Electric & Power Company	06/30/98	Transnuclear West, Incorporated	Metal Cask TN-32

*Plant undergoing decommissioning

Table 2

NRC dry spent fuel storage licensees

RISKS AND ALTERNATIVE OPTIONS
ASSOCIATED WITH SPENT FUEL STORAGE AT THE
SHEARON HARRIS NUCLEAR POWER PLANT

Appendix A

Spent fuel management at the Harris plant

1. Introduction

This appendix summarizes present and proposed arrangements for managing spent fuel at the Shearon Harris plant. Carolina Power & Light Company (CP&L), the licensee for the plant, proposes to introduce new arrangements for spent fuel management. For that purpose, CP&L seeks an amendment to the plant's operating license. Unless specified otherwise, information presented here is drawn from CP&L's application to amend the Harris license, from CP&L's Final Safety Analysis Report (FSAR) for the Harris plant, or from viewgraphs shown by CP&L personnel during meetings with staff of the Nuclear Regulatory Commission (NRC).[1]

2. Present and proposed spent fuel storage capacity

The Harris plant features one pressurized-water reactor (PWR). The core of this reactor contains 157 fuel assemblies, with a center-center distance of about 8.5 inches. The Harris plant was to have four units but only the first unit was built. (A unit consists of a reactor, a turbine-generator and associated equipment.) A fuel handling building was built to serve all four units. This building contains four fuel pools (A, B, C, D), a cask loading pool and three fuel transfer canals, all interconnected but separable by gates. Figure A-1 shows a plan view of the interior of the fuel handling building.

Pools A and B

Pools A and B contain fuel racks, and are in regular use. CP&L says that fresh fuel, and spent fuel recently discharged from the Harris reactor, is stored in pool A. Fuel examination and repair are performed in an open space in pool

[1] Meetings between NRC staff and CP&L representatives, to discuss the proposed license amendment, were held on 3 March 1998 and 16 July 1998.

B. Pools C and D are flooded but do not contain racks. The cooling and water cleanup systems for pools C and D were never completed.

Pool A now contains six racks (360 fuel assembly spaces) for PWR fuel and three racks (363 spaces) for boiling-water reactor (BWR) fuel, for a total pool capacity of 723 fuel assemblies. Pool B contains twelve PWR racks (768 spaces) and seventeen BWR racks (2,057 spaces), and is licensed to store one additional BWR rack (121 spaces), for a total, potential pool capacity of 2,946 fuel assemblies. Thus, pools A and B now have a combined, potential capacity of 3,669 fuel assemblies. The center-center distance in the racks in pools A and B is 10.5 inches for PWR fuel and 6.25 inches for BWR fuel.

Pools A and B store spent fuel from the Harris reactor and from CP&L's Brunswick plant and Robinson plant. The Brunswick plant has two BWRs while the Robinson plant has one PWR. Shipment of spent fuel from Brunswick and Robinson to Harris is said by CP&L to be necessary to allow core offload capacity in the pools at Brunswick and Robinson.

Pools C and D

CP&L seeks an amendment to its operating license so that it can activate pools C and D at Harris. By activating these pools, CP&L expects to have sufficient storage capacity at its three nuclear plants to accommodate all the spent fuel discharged by the four CP&L reactors (the Harris and Robinson PWRs and the two Brunswick BWRs) through the ends of their current operating licenses.

CP&L plans to install racks in pool C in three campaigns (approximately in 2000, 2005 and 2014), to create 927 PWR spaces and 2,763 BWR spaces, for a total capacity in this pool of 3,690 fuel assemblies. Thereafter, CP&L plans to install racks in pool D in two campaigns (approximately in 2016 and at a date to be determined), to create 1,025 PWR spaces. Thus, the ultimate capacity of pools C and D will be 4,715 fuel assemblies. The center-center distance in the racks used in these pools will be 9.0 inches for PWR fuel and 6.25 inches for BWR fuel.

The PWR racks in pools C and D have a smaller center-center distance than the racks in pools A and B (9.0 inches instead of 10.5 inches). This arrangement allows more PWR fuel to be placed in a given pool area but also means that PWR fuel in pools C and D is more prone to undergo criticality. In response, CP&L proposes to include in the Technical Specifications for Harris a provision that PWR fuel will not be placed in pools C and D unless it has relatively low enrichment and high burnup.[2]

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[2] License amendment application, Enclosure 5.

Summary

Table A-1 summarizes the present and proposed storage capacity in the Harris pools. At present, pools A and B have a combined, potential capacity of 3,669 assemblies. The proposed, combined capacity of pools C and D will be 4,715 assemblies. Thus, activation of pools C and D will represent an increase of about 130 percent in the number of fuel assemblies that could be stored at Harris.

3. Support services for pools C and D

The water in a spent fuel pool must be cooled and cleaned. Figure A-2 provides a schematic view of typical cooling and cleanup systems. It will be noted that pool water is circulated through heat exchangers, where its heat is transferred to a secondary cooling system. At Harris, the secondary cooling system is the component cooling water (CCW) system. Water in the secondary system is in turn circulated through heat exchangers, where its heat is transferred to a tertiary cooling system. At Harris, the tertiary cooling system is the service water (SW) system.

When the Harris plant was designed, the intention was that pools C and D would be cooled by the CCW system for the second unit. That unit was never built and its CCW system does not exist. Thus, CP&L plans to cool pools C and D by completing their partially built cooling systems and connecting those systems to the CCW system of the first unit. The Unit 1 CCW system already provides cooling to pools A and B and serves other, important safety functions. For example, the Unit 1 CCW system provides cooling for the residual heat removal (RHR) system and reactor coolant pumps of the Unit 1 reactor.

The original design concept for Harris

In the Harris plant's original design concept, pools A and B would have served Units 1 and 4, while pools C and D would have served Units 2 and 3. There would have been a separate, fully-redundant, 100 percent-capacity cooling and water cleanup system for each pair of pools (A+B and C+D). Cooling of pools C and D would have been provided by the CCW system of Unit 2. Electrical power for the pumps that circulate water from the C and D pools through heat exchangers (see Figure A-2) would have been supplied by the Unit 2 electrical systems. Pools A and B would have been supported by the CCW and electrical systems of Unit 1.

During CP&L's planning for the activation of pools C and D, the company considered the construction of an independent system to cool these pools. Within that option, CP&L considered the further possibility of providing dedicated emergency diesel generators to meet the electrical needs of pools C and D if normal electricity supply were unavailable. Construction of an independent cooling system for pools C and D, supported by dedicated emergency diesel generators, could provide the level of safety that was associated with the original design concept for Harris. However, CP&L has not proceeded with this option.

Capacity of the Unit 1 CCW system

According to CP&L's license amendment application, the bounding heat load from the fuel in pools C and D will be 15.6 million BTU/hour (4.6 MW).[3] At present, the Unit 1 CCW system cannot absorb this additional heat load. Thus, CP&L proposes to include in the Technical Specifications for Harris an interim provision that the heat load in pools C and D will not be allowed to exceed 1.0 million BTU/hour.[4] CP&L claims that an additional heat load of 1.0 million BTU/hour can be accommodated by the Unit 1 CCW system, and that the fuel to be placed in pools C and D will not create a heat load exceeding 1.0 million BTU/hour through 2001.

CP&L contemplates a future upgrade of the Unit 1 CCW system, so that this system can accommodate an additional heat load of 15.6 million BTU/hour from pools C and D. This contemplated upgrade is not described in the present license amendment application. Apparently, CP&L intends to perform the upgrade of the Unit 1 CCW system concurrent with a power uprate for the Unit 1 reactor. A 4.5 percent power uprate of the reactor will be associated with steam generator replacement, and will take effect in about 2002. About two years later, there will be a further power uprate of 1.5 percent. CP&L projects that the Unit 1 CCW heat load, including the reactor power uprate and the ongoing use of pools C and D, will substantially exceed the capability of the present CCW system.

To summarize, CP&L's short-term plan (through 2001) for cooling pools C and D is to exploit the margin in the Unit 1 CCW system, so as to accommodate an additional heat load of 1.0 million BTU/hour. CP&L's longer-term plan is to upgrade the CCW system, in a manner not yet specified, so as to accommodate an additional heat load of 15.6 million BTU/hour. The CCW upgrade must also accommodate an increase in the rated power of the Harris reactor. CP&L expects that the design of the CCW

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[3] License amendment application, Enclosure 7, page 5-16.
[4] License amendment application, Enclosure 5.

upgrade will commence in mid-1999 and will be completed in early 2001, one year after the company expects pool C to enter service.

Safety implications

In order to exploit the margin in the existing CCW system so as to cool pools C and D, CP&L may be obliged to require its operators to divert some CCW flow from the RHR heat exchangers during the recirculation phase of a design-basis loss-of-coolant accident (LOCA) event at the Harris reactor.[5] This is a safety issue because, during the recirculation phase of a LOCA, operation of the RHR system is essential to keeping the reactor core and containment in a safe condition. CP&L's exploitation of the margin in the existing CCW system is deemed by CP&L and NRC to constitute an "unreviewed safety question".[6]

In Enclosure 9 of its license amendment application, CP&L provides a brief description of the analysis that is has performed to demonstrate that an additional load of 1.0 million BTU/hour is within the marginal capacity of the Unit 1 CCW system. That analysis is said by CP&L to take the form of a 10CFR50.59 Safety Evaluation. The description in Enclosure 9 raises more questions than it answers, and does not address the practical issues that affect an analysis of a cooling system's thermal margin. For example, CP&L has mentioned elsewhere that exploitation of the margin in the Unit 1 CCW system could involve changes in design assumptions that include fouling factors and tube plugging limits.[7] These matters are not addressed in Enclosure 9.

As background, note that the Unit 1 CCW system has two heat exchangers, each with a design heat transfer rate of 50 million BTU/hour. During the recirculation phase of a design-basis LOCA, the estimated maximum heat load to be extracted from the CCW system by the SW system is 160 million BTU/hour.[8] These numbers suggest that accommodating a design-basis LOCA will already exploit the margin of the CCW system, without any additional load from pools C and D.

Lack of QA documentation

Activation of pools C and D will require the completion of their cooling and water cleanup systems, and the connection of their cooling systems to the

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[5] License amendment application, Enclosure 9.
[6] Ibid; Federal Register: January 13, 1999 (Volume 64, Number 8), pages 2237-2241.
[7] Viewgraphs for presentation by CP&L to the NRC staff, 3 March 1998.
[8] Harris FSAR, section 9.2, Amendment No. 40.

existing CCW system. CP&L states that approximately 80 percent of the necessary piping was completed before the second Harris reactor was cancelled.[9] However, some of the quality assurance (QA) documentation for the completed piping is no longer available. Much of the completed piping is embedded in concrete and is therefore difficult or impossible to inspect. To address this situation, CP&L proposes an Alternative Plan to demonstrate that the previously completed piping and other equipment is adequate for its purpose.[10] Nevertheless, the cooling systems for pools C and D will not satisfy ASME code requirements.

Electrical power

The cooling systems for pools C and D will draw electrical power from the electrical systems of Unit 1. If electricity supply to the cooling pumps for pools C and D is interrupted, the pools will heat up and eventually boil. CP&L says that pools C and D will begin to boil after a time period "in excess of 13 hours", assuming a bounding decay heat load of 15.6 million BTU/hour.[11] To prevent the onset of pool boiling in the event of a loss of offsite power, the Harris operators may be obliged to provide electrical power to pools C and D from the existing emergency diesel generators, which also serve pools A and B and the Unit 1 reactor. In its license amendment application, CP&L does not address the ability of the emergency diesel generators to meet the additional electrical loads associated with pools C and D. CP&L does mention in the Harris FSAR the potential for connecting "portable pumps" to bypass the pool cooling pumps should the latter be inoperable.[12] However, the characteristics, capabilities and availability of such portable pumps are not addressed in the license amendment application.

4. Potential cesium-137 inventory of the Harris pools

For the purposes of Appendix E of this report, it is necessary to estimate the potential inventory of the radioisotope cesium-137 in the Harris pools. As a starting point, consider the inventory of cesium-137 in a typical PWR spent fuel assembly, represented here by an average assembly in batch 16 from the Ginna plant, discharged in April 1987. At discharge, the Ginna assembly contained 1.4×10^5 Curies of cesium-137 per metric ton of heavy metal (MTHM).[13]

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[9] License amendment application, Enclosure 1, page 4.
[10] License amendment application, Enclosure 8.
[11] License amendment application, Enclosure 7, page 5-8.
[12] Harris FSAR, page 9.1.3-6, Amendment No. 48.
[13] V L Sailor et al, Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82, NUREG/CR-4982, July 1987, Appendix A.

A Harris PWR assembly has a mass of 0.461 MTHM. Thus, one can estimate that a typical Harris assembly contains, at discharge, 0.65 x 10^5 Curies of cesium-137. The assembly's content of cesium-137 will decline exponentially, with a half-life of 30 years. At the same age after discharge, a typical BWR assembly in the Harris pools will contain about 1/4 of the amount of cesium-137 in a Harris PWR assembly.[14]

Potential stock of assemblies in the Harris pools

Table A-2 shows CP&L's projection of the stock of assemblies in Harris pools C and D, for the purposes of bounding analysis. A CP&L representative has stated that CP&L will not ship fuel to Harris until it has aged for 3 years, and will not place fuel in pools C and D until it has aged for 5 years.[15] Accepting that fuel aged less than 3 years will not be shipped to Harris, one can assume, to supplement Table A-2, that the Harris pools will contain 456 BWR assemblies aged for 3 years, 172 PWR assemblies aged for 3 years, and 96 PWR assemblies aged for 1 year. Hereafter, these assumptions and Table A-2 are taken to represent the potential stock of fuel assemblies in the Harris pools.

On this basis, the Harris pools' stock of spent fuel aged 3 years or less will be 268 PWR assemblies and 456 BWR assemblies. All of this fuel might be in pools A and B, although there is nothing in CP&L's present or proposed Technical Specifications which prohibits placement of recently discharged fuel in pools C and D. On the same basis, the Harris pools' stock of spent fuel aged 9 years or less will be 784 PWR assemblies and 1,824 BWR assemblies.

Inventory of cesium-137

Now consider the inventory of cesium-137 in the Harris pools. Assume that a newly discharged PWR assembly contains 0.65 x 10^5 Curies of cesium-137, neglect the difference between Harris and Robinson assemblies, allow for radioactive decay, and assume that a BWR assembly contains 1/4 of the amount of cesium-137 in a PWR assembly of the same age. Then, the Harris pools' stock of spent fuel aged 3 years or less will contain 2.3 x 10^7 Curies (870,000 TBq) of cesium-137, with a mass of 260 kilograms. Also, the Harris pools' stock of spent fuel aged 9 years or less will contain 7.1 x 10^7 Curies (2,600,000 TBq) of cesium-137, with a mass of 790 kilograms.

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[14] The ratio of 1/4 derives from the parameters shown in the license amendment application, Enclosure 7, page 5-15.
[15] J Scarola of CP&L, presentation to Orange County Board of Commissioners, 9 February 1999.

CP&L could provide a more precise projection of the cesium-137 inventory in the Harris pools over coming years. However, our estimate will be a reasonable indication of cesium-137 inventory during the next two decades, assuming pools C and D are used as CP&L intends.

For comparison with the pools' inventory of cesium-137, note that the NRC has estimated the inventory of cesium-137 in the Harris reactor core, during normal operation, to be 4.2×10^6 Curies (155,000 TBq, or 47 kilograms).[16] This represents an average inventory of 0.27×10^5 Curies in each of the reactor's 157 fuel assemblies. Note that an average assembly in the core will have a lower cesium-137 content than an assembly at discharge, and that the NRC's estimate may have assumed a relatively low fuel burnup.

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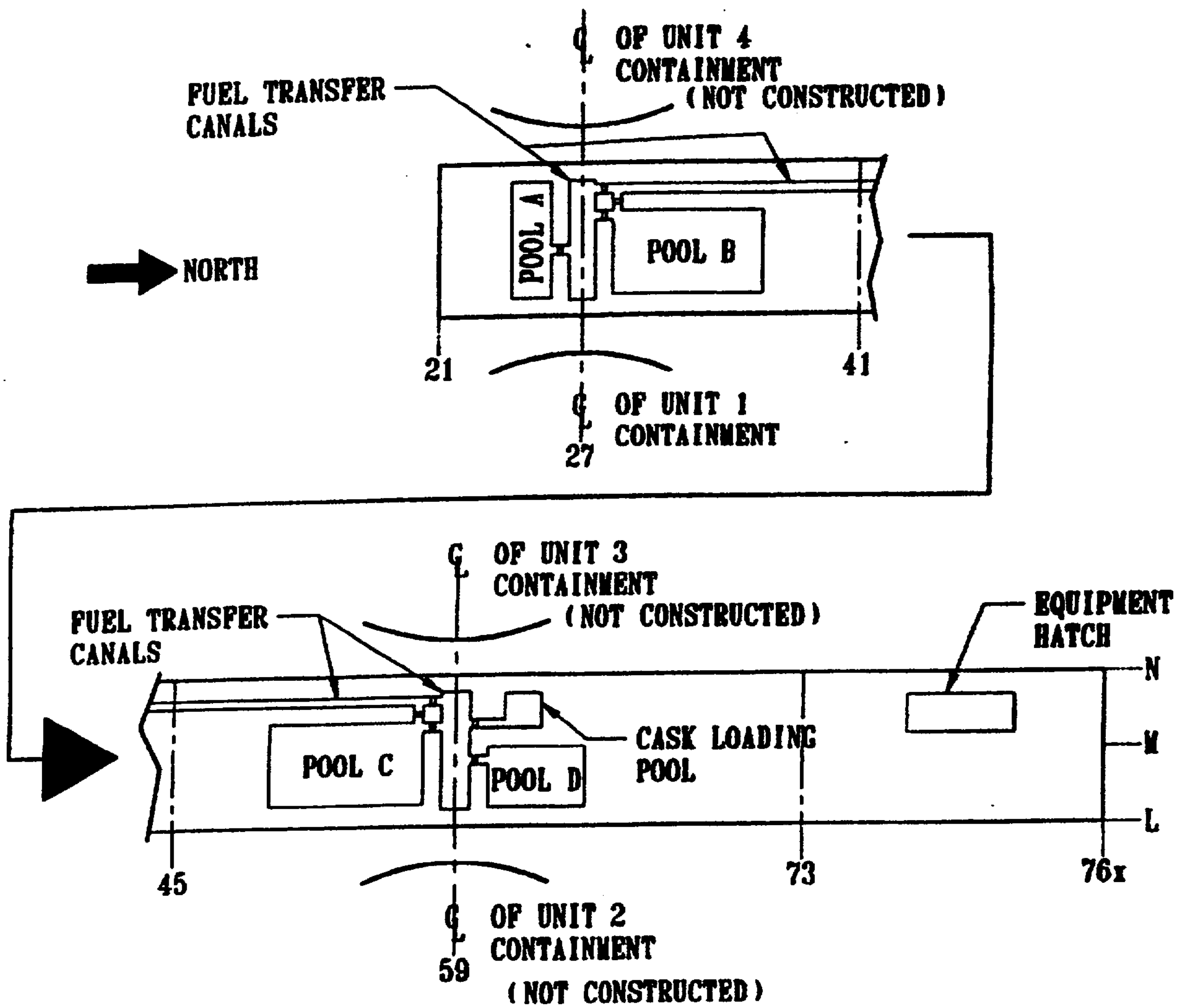
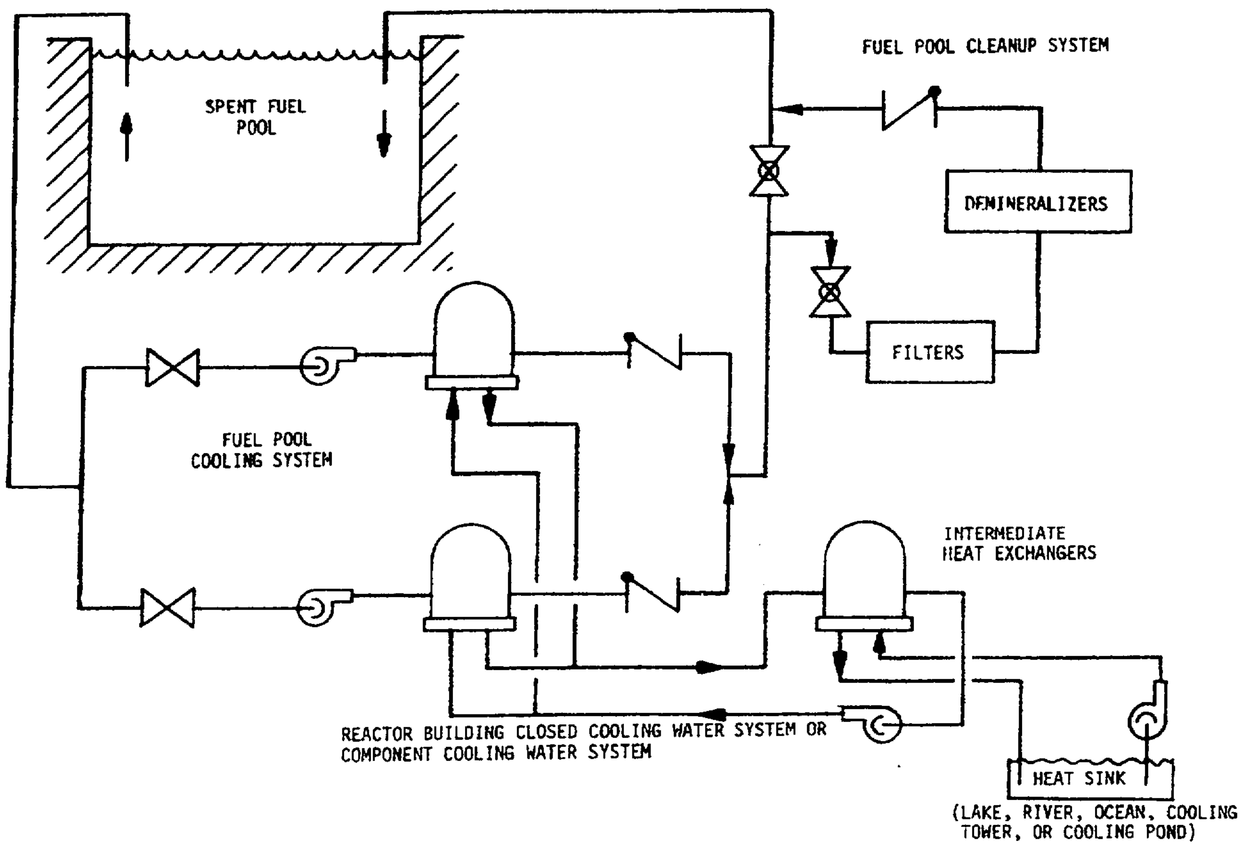
[16] US Nuclear Regulatory Commission, Final Environmental Statement Related to the Operation of Shearon Harris Nuclear Power Plant Units 1 and 2, NUREG-0972, October 1983.



Source: License amendment application

Figure A-1

Interior of the Harris Fuel Handling Building



Source: NUREG-0404

Figure A-2

Typical cooling and cleanup systems for a spent fuel pool

Pool	PWR spaces	BWR spaces	Total
'A'	360	363	723
'B'	768	2178	2946
'C'	927	2763	3690
'D'	1025	0	1025
Total	3080	5304	8384

Source: License amendment application

Table A-1

Present and proposed storage capacity in the Harris pools

DECAY PERIODS FOR A BOUNDING POOLS C AND D STORAGE CONFIGURATION			
PWR Fuel Assemblies		**BWR Fuel Assemblies**	
Number of Assys	Decay Period	Number of Assys	Decay Period
172	5 years	456	5 years
172	7 years	456	7 years
172	9 years	456	9 years
172	11 years	456	11 years
172	13 years	456	13 years
172	15 years	483	15 years
172	17 years		
172	19 years		
172	21 years		
172	23 years		
232	25 years		

Source: License amendment application

Table A-2

Projected stock of fuel assemblies in Harris pools C and D

RISKS AND ALTERNATIVE OPTIONS
ASSOCIATED WITH SPENT FUEL STORAGE AT THE
SHEARON HARRIS NUCLEAR POWER PLANT

Appendix B

Potential for severe accidents at the Harris reactor

1. Introduction

In examining the risks associated with spent fuel storage at Harris, one must consider the potential for accidents at the Harris reactor. Such consideration is necessary for two reasons. First, a reactor accident could accompany, initiate or exacerbate a spent fuel pool accident. Second, modification of the Harris plant to increase its spent fuel storage capacity could increase the probability or consequences of accidents at the Harris reactor.

This appendix addresses the potential for severe accidents at the Harris reactor. "Severe" reactor accidents have two major defining characteristics. First, they involve substantial damage to the reactor core, with a corresponding release of radioactive material from the fuel assemblies. Second, they extend the envelope of potential accidents beyond the "design basis" accidents that were considered when US reactors were first licensed.

During a severe reactor accident, radioactive material may be released to the environment, as an atmospheric plume or by entry into ground or surface waters. The release may be large or small. In illustration, the 1979 TMI accident and the 1986 Chernobyly accident were both severe accidents, involving substantial damage to the reactor core. However, the TMI release was comparatively small and the Chernobyl release was comparatively large.

2. Probabilistic risk assessment

The probabilities and consequences of potential accidents at nuclear facilities can be estimated through the techniques of probabilistic risk assessment (PRA). Nuclear facility PRAs are performed at three levels. At Level 1, a PRA will estimate the probability of a specified type of accident (e.g., severe core damage at a reactor). At Level 2, which builds upon Level 1 findings, a PRA will estimate the nature of potential radioactive releases from the facility. In turn, the Level 2 findings can be used in a Level 3 exercise, which will estimate the offsite consequences (health effects, economic effects, etc.) of radioactive releases. For all three levels, a PRA can be performed for "internal" accident-initiating events (equipment failure, operator error, etc.) and for "external" accident-initiating events (earthquakes, floods, etc.).[1]

PRA methodology is used for non-reactor nuclear facilities, but is most highly developed in its application to reactors. The first PRA was the Reactor Safety Study (WASH-1400), which was published by the US Nuclear Regulatory Commission (NRC) in 1975.[2] The present state of the PRA art is exemplified by a study of five nuclear power plants (NUREG-1150) published by the NRC in 1990.[3]

Uncertainty and incompleteness of PRA findings

An in-depth PRA such as NUREG-1150 can provide useful insights regarding a reactor's accident potential. However the findings of any PRA will inevitably be accompanied by substantial uncertainty and incompleteness. Uncertainty arises from the intrinsic difficulties of modelling complex systems, and from limited understanding of some of the physical processes that accompany severe accidents. Incompleteness arises from the potential for unanticipated accident sequences, gross human errors, undetected structural flaws, and acts of malice or insanity.[4] Thus, a PRA's finding about the probability of an accident should be viewed with two caveats. First, the accident probability, as found in the PRA, will fall within some range of uncertainty. Second, the accident probability, as found in the PRA, will be a lower bound to the true probability, which will be impossible to determine.

NUREG-1150 findings for the Surry PWRs

Figures B-1 and B-2 illustrate the findings of NUREG-1150. These figures show the estimated core damage frequency for the Surry nuclear reactors. These reactors are 3-loop Westinghouse pressurized-water reactors (PWRs), as is the Harris reactor. Core damage frequency is shown per reactor-year of

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1 In PRA practice, it is common for analysis of externally-initiated accidents to build upon previous analysis of internally-initiated accidents.
2 US Nuclear Regulatory Commission, Reactor Safety Study, WASH-1400 (NUREG-75/014), October 1975.

3 US Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five US Nuclear Power Plants, NUREG-1150 (2 vols), December 1990.
4 H Hirsch, T Einfalt, O Schumacher and G Thompson, IAEA Safety Targets and Probabilistic Risk Assessment, Gesellschaft fur Okologische Forschung und Beratung, Hannover, August 1989.

operation. Figure B-1 shows core damage frequency for internal events, fires and earthquakes (seismic events). Two estimates are shown for seismic events, one drawing on an estimate of earthquake frequency by Lawrence Livermore National Laboratory, the other on an estimate by the Electric Power Research Institute (EPRI). The bars in Figure B-1 span an estimated uncertainty range from the 5th to the 95th percentile. An alternative portrayal of estimated uncertainty is provided by the probability densities shown in Figure B-2.

The authors of NUREG-1150 made a considerable effort to estimate the uncertainty associated with their findings. However, their uncertainty estimates relied heavily on expert opinion, rather than on a statistical analysis of data. Thus, the uncertainty estimates in NUREG-1150 should be viewed with caution. The reader will observe a cautionary statement attached to Figures B-1 and B-2. Finally, the NUREG-1150 findings of accident probability must be viewed as lower bounds, as explained above.

Acts of malice

Nuclear reactor PRAs do not consider malicious acts such as sabotage, terrorism or acts of war. Such acts are less susceptible to probabilistic analysis than are accident initiators such as human error. Nevertheless, sabotage and terrorism pose a significant threat to US nuclear plants.[5] NRC regulations oblige reactor licensees to take certain precautions against this threat, but these precautions do not preclude the possibility of successful acts of sabotage or terrorism.

The US government is increasing the level of attention and the expenditure that it devotes to the threat of terrorism. Many observers argue that greater effort is required. For example, three authors with high-level government experience have recently written:[6]

> Long part of the Hollywood and Tom Clancy repertory of nightmarish scenarios, catastrophic terrorism has moved from far-fetched horror to a contingency that could happen next month. Although the United States still takes conventional terrorism seriously, as demonstrated by the response to the attacks on its embassies in Kenya and Tanzania in August, it is not yet prepared for the new threat of catastrophic terrorism.

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[5] G Thompson, War, Terrorism and Nuclear Power Plants, Peace Research Centre, Australian National University, October 1996.
[6] A Carter, J Deutch and P Zelikow, "Catastrophic Terrorism", Foreign Affairs, November/December 1998, page 80.

The effectiveness of licensees' arrangements to resist terrorist attacks on nuclear plants has recently been a subject of public debate. According to the head of the NRC's Operational Safeguards Response Evaluation program, plant security arrangements have failed in at least 14 of the 57 mock assaults which the NRC has conducted since 1991. Nevertheless, the NRC intends to weaken its oversight of licensees' antiterrorism efforts.[7]

3. The Harris IPE and IPEEE

The NRC requires each holder of a reactor license to perform an Individual Plant Examination (IPE), to assess the severe accident potential of that reactor. Carolina Power and Light (CP&L) submitted an IPE for the Harris reactor in 1993.[8] This was a Level 2 PRA for internal events, including in-plant flooding but neglecting in-plant fires.

The NRC also requires each licensee to perform an Individual Plant Examination for External Events (IPEEE). CP&L submitted an IPEEE for the Harris reactor in 1995.[9] This study did not follow PRA practice. Instead, it consisted of a seismic margins analysis and a limited analysis of in-plant fires.

IPE estimate of core damage frequency

According to the IPE performed by CP&L, the frequency of severe core damage at Harris is 7×10^{-5} per reactor-year. This must be considered a "point" estimate, because the Harris IPE does not provide an uncertainty band or probability density function of the kind shown in Figures B-1 and B-2. The IPE predicts that accident sequences involving a loss-of-coolant accident (LOCA) will account for 40 percent of Harris' core damage frequency, while sequences involving station blackout (loss of electrical power) will account for 26 percent of the core damage frequency. The 40 percent contribution of LOCAs to core damage frequency is due to LOCAs with injection failure (17 percent) and LOCAs with recirculation failure (23 percent).

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[7] S Allen, "NRC to cut mock raids on atom plants", The Boston Globe, 25 February 1999, page A6.
[8] Carolina Power & Light Company, Shearon Harris Nuclear Power Plant, Unit No. 1: Individual Plant Examination Submittal, August 1993.
[9] Carolina Power & Light Company, Shearon Harris Nuclear Power Plant, Unit No. 1: Individual Plant Examination for External Events Submittal, June 1995.

The NRC has compiled and compared IPE findings for all US commercial nuclear reactors.[10] Some of the results are shown in Figures B-3 and B-4. Figure B-3 shows that the reported core damage frequencies tend to be significantly higher for PWRs than for boiling-water reactors (BWRs). Figure B-4 shows that the reported core damage frequencies tend to be higher for 3-loop Westinghouse (W-3) PWRs than for 2-loop and 4-loop Westinghouse PWRs and PWRs made by Combustion Engineering (CE) and Babcock & Wilcox (B&W). The Harris reactor is a 3-loop Westinghouse PWR.

From its compilation of IPE findings, the NRC concluded that sequences involving LOCAs (especially LOCAs with recirculation failure) and station blackout are major contributors to estimated core damage frequency at 3-loop Westinghouse PWRs. This conclusion is consistent with the Harris IPE findings outlined above. The NRC noted that the 3-loop Westinghouse PWRs exhibit a relatively high dependence of front-line safety systems on service water (SW), component cooling water (CCW) and heating, ventilating & air conditioning (HVAC) systems.

IPEEE findings

The Harris IPEEE consisted of a seismic margins analysis and a limited analysis of in-plant fires. The seismic margins analysis examined the Harris reactor's ability to withstand a review level eathquake (RLE) of 0.3g. Note that the reactor's safe shutdown earthquake (SSE) is 0.15g and its operating basis earthquake is 0.075g. According to the IPEEE, the only actions required to make the Harris reactor safe against the RLE involved housekeeping and minor modifications, and these actions have been taken. The IPEEE did not investigate the implications of an earthquake more severe than the RLE.

A limited analysis of in-plant fires appears in the IPEEE. This analysis identified four fire scenarios as significant contributors to core damage frequency. One scenario would take place in each of switchgear rooms A and B, and two scenarios would take place in the control room. The combined core damage frequency, summed over all four scenarios, would be 1×10^{-5} per reactor-year, but the IPEEE argues that a summation of this kind would be inaccurate without further refinement of the analysis.

Figures B-1 and B-2 illustrate the findings that can be generated by the systematic application of PRA techniques to accident sequences initiated by external events. In comparison, the Harris IPEEE is a relatively crude study.

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[10] US Nuclear Regulatory Commission, Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, NUREG-1560 (3 vols), December 1997.

Release of radioactive material

The Harris IPE analyzes the potential for accident sequences to release radioactive material to the environment. The IPE only considers releases to the atmosphere during accident sequences that are initiated by internal events. Potential releases are described by a set of release categories.

Release category RC-5 represents the largest release identified in the IPE. This release would include 100 percent of the noble gas inventory in the reactor core, 59 percent of the CsI inventory, and 53 percent of the CsOH inventory. The IPE does not describe how cesium would be distributed between CsI and CsOH. Thus, one can interpret the RC-5 release as including 59 percent of iodine isotopes in the core and 53-59 percent of cesium isotopes.

Accident sequences contributing to release category RC-5 would involve steam generator tube rupture (SGTR) with a stuck-open safety relief valve (SRV), or an inter-system LOCA (ISLOCA). The SGTR could occur as an accident initiating event or through overheating of steam generator tubes during an accident sequence initiated by some other event. A stuck-open SRV, concurrent with a SGTR, would create a direct pathway from the reactor core to the atmosphere, bypassing the containment. In an ISLOCA sequence, reactor cooling water would be lost from a breach in a piping system outside the containment. This loss of water would initiate the accident, and the water's escape pathway would provide a route for the escape of radioactivity after core damage began.

An accident in release category RC-5 would cause substantial offsite exposure to radioactivity. In addition, the Harris plant and its immediate surroundings would become radioactively contaminated to the point where access by personnel would be precluded. Accidents in other release categories would release smaller amounts of radioactive material, but could also contaminate the Harris plant to the point where access by personnel would be precluded. This matter is addressed further in Appendix C.

The Harris IPE estimates the probability of release category RC-5 as 3×10^{-6} per reactor-year. Note that the overall probability of core damage is estimated to be 7×10^{-5} per reactor-year. Thus, the IPE predicts that 4 percent of core damage sequences would yield a release in category RC-5. Overall, the IPE predicts that 15 percent of core damage sequences would be accompanied by a significant degree of containment failure or bypass, with a total probability of about 1 x 10^{-5} per reactor-year.[11]

4. Pool-reactor interactions

Neither CP&L nor NRC have performed an analysis to determine how a severe accident or a design-basis accident at the Harris reactor might accompany, initiate or exacerbate an accident at the Harris fuel pools, or vice versa.[12] Appendix C shows how a severe reactor accident could initiate a pool accident by precluding personnel access. From Appendix E it can be inferred that a pool accident could similarly preclude access to the reactor.

The Harris IPE does not analyze the implications that activation of pools C and D at Harris might have for severe accidents at the Harris reactor. Appendix A points out that activation of pools C and D will raise two safety issues that could increase the probability of core damage at Harris. First, cooling of pools C and D and a planned uprate in reactor power will place an increased heat load on the component cooling water (CCW) system of Harris Unit 1, thus adding stress to operators and equipment at Harris, potentially increasing the probability of core damage. Second, cooling of pools C and D will create an increased load on the electrical systems at Harris, thereby adding stress to operators and equipment and potentially increasing the probability of core damage. Before activation of pools C and D is permitted, these effects should be examined through a supplement to the Harris IPE.

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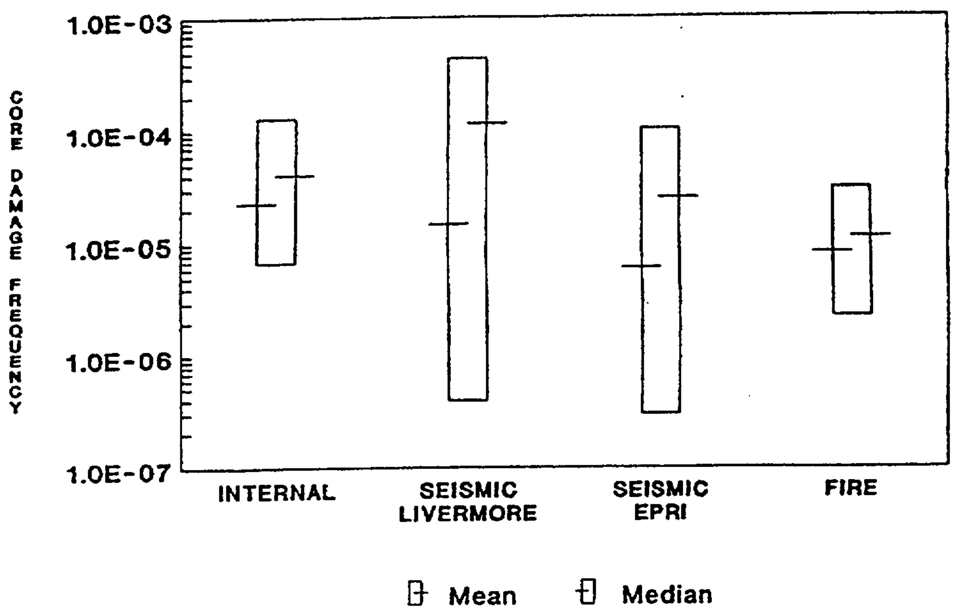
[11] Release categories involving significant containment failure or bypass are, in descending order of estimated probability, RC-4, RC-5, RC-6, RC-1B, RC-4C and RC-3. Each of these categories involves a 100 percent release of noble gases. The CsI release fraction ranges from .001 percent (RC-6) to 59 percent (RC-5).

[12] As examples of literature relevant to potential safety interactions between fuel pools and reactors, see: D A Lochbaum, Nuclear Waste Disposal Crisis, PennWell Books, Tulsa, OK, 1996; and N Siu et al, Loss of Spent Fuel Pool Cooling PRA: Model and Results, INEL-96/0334, Idaho National Engineering Laboratory, September 1996.

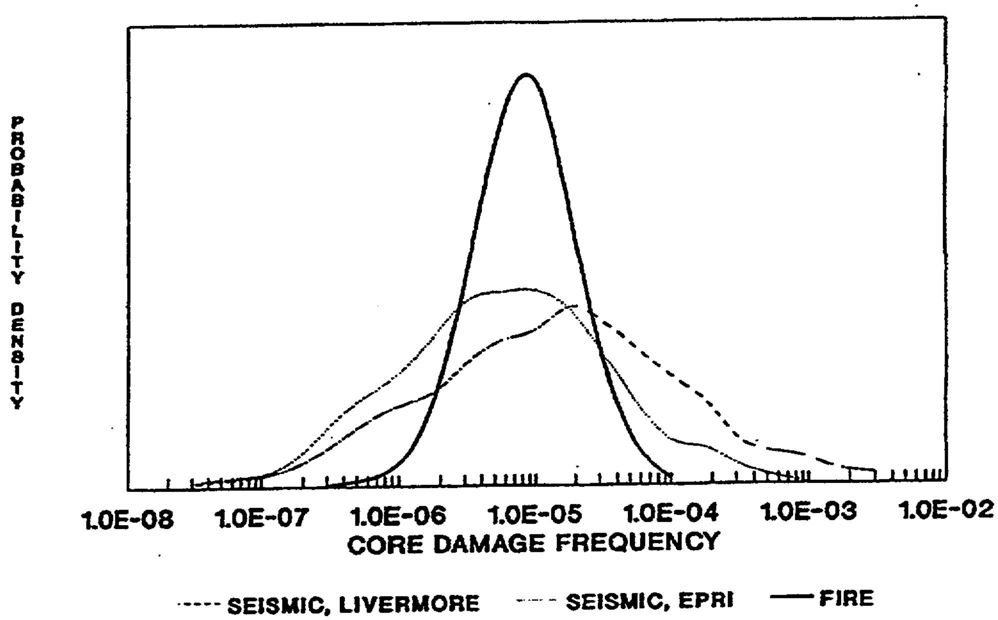


Note: As discussed in Reference 8.7, core damage frequencies below 1E–5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Source: NUREG-1150

Figure B-1

Estimated core damage frequency for the Surry PWRs

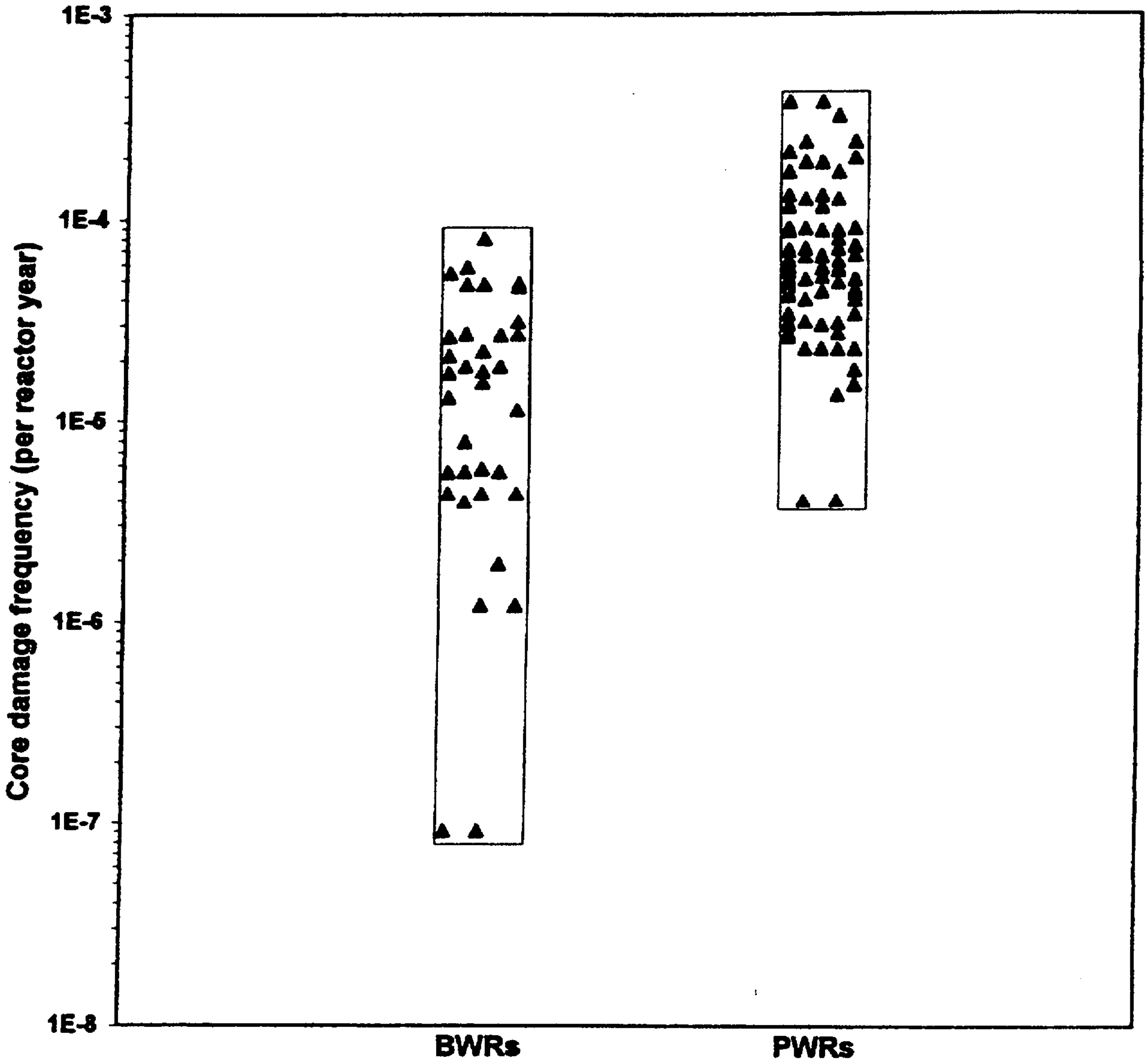


Note: As discussed in Reference 8.7, core damage frequencies below 1E–5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Source: NUREG-1150

Figure B-2

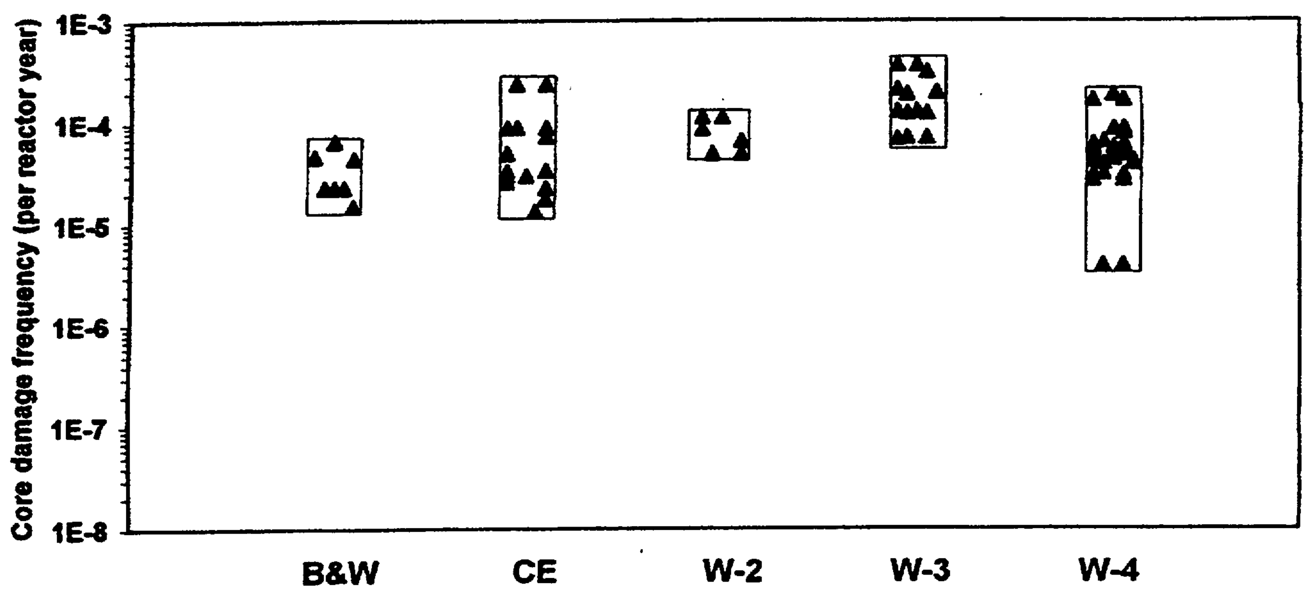
Probability density of estimated external-events core damage frequency for the Surry PWRs



Source: NUREG-1560

Figure B-3

Summary of core damage frequencies as reported in IPEs



Source: NUREG-1560

Figure B-4

Core damage frequencies reported in IPEs for types of PWR

RISKS AND ALTERNATIVE OPTIONS ASSOCIATED WITH SPENT FUEL STORAGE AT THE SHEARON HARRIS NUCLEAR POWER PLANT

Appendix C

Potential for loss of water from the Harris pools

1. Introduction

This appendix considers the potential for partial or total loss of water from one or more of the Harris fuel pools. The arrangement and use of these pools are described in Appendix A. If a loss of water occurs, then exothermic reactions could occur in the affected pools, as described in Appendix D.

2. Types of event that might cause water loss

A variety of events, alone or in combination, might lead to partial or complete uncovering of spent fuel in the Harris pools. Relevant types of event include:

(a) an earthquake, cask drop, aircraft crash, human error, equipment failure or sabotage event that leads to direct leakage from the pools;
(b) siphoning of water from the pools through accident or malice;
(c) interruption of pool cooling, leading to pool boiling and loss of water by evaporation; and
(d) loss of water from active pools into adjacent pools or canals that have been gated off and drained.

3. Assessing the potential for water loss: the role of PRA

A discipline known as probabilistic risk assessment (PRA) has been developed to examine the probabilities and consequences of potential accidents at nuclear facilities. PRA techniques are most highly developed in their application to reactor accidents, but can be applied to fuel pool accidents. Appendix B describes the characteristics, strengths and limitations of PRA.

Carolina Power & Light Company (CP&L) has prepared a Level 2, internal-events PRA for the Harris reactor, in the form of an Individual Plant

Examination (IPE). CP&L has also performed a limited assessment of the vulnerability of the Harris reactor to earthquakes and in-plant fires, in the form of an Individual Plant Examination for External Events (IPEEE). The findings of the IPE and IPEEE are described in Appendix B.

The Harris IPE and IPEEE could be extended to encompass fuel pool accidents as well as reactor accidents. Such an extension would be logical, because there are various ways in which a severe accident or a design-basis accident at the Harris reactor might accompany, initiate or exacerbate an accident at the Harris fuel pools, or vice versa.[1] However, there is no current indication that CP&L will extend the IPE or IPEEE, or will otherwise apply PRA techniques to potential accidents at the Harris fuel pools.

As an indication of the need for an extended IPE and IPEEE at Harris, covering fuel pool accidents, consider a study performed for the NRC by analysts at the Idaho National Engineering Laboratory.[2] These analysts examined a two-unit boiling-water reactor (BWR) plant based on the Susquehanna plant. They estimated that the plant's probability of spent fuel pool (SFP) boiling events is 5×10^{-5} per year From Appendix B it will be noted that the Harris IPE predicts a core damage frequency of 7×10^{-5} per year. (Years and reactor-years are equivalent for Harris.) The similar magnitudes of these probabilities suggests that pool accidents could be a major contributor to risk at Harris, especially considering the large inventory of long-lived radioisotopes in the Harris pools.

A comprehensive application of PRA techniques to the Harris fuel pools is a task beyond the scope of the author's present work for Orange County. In the remainder of this appendix, selected issues are discussed. These discussions illustrate the need for a comprehensive PRA approach.

4. Analyses of earthquake and cask drop at the Robinson plant

Analysts sponsored by the Nuclear Regulatory Commission (NRC) have examined the effects of a severe earthquake and a cask drop on the fuel pool at CP&L's Robinson plant.[3] The Robinson plant features one pressurized-water reactor (PWR) and a single fuel pool. By examining the vulnerability of

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[1] As examples of literature relevant to potential safety interactions between fuel pools and reactors, see: D A Lochbaum, Nuclear Waste Disposal Crisis, PennWell Books, Tulsa, OK, 1996; and N Siu et al, Loss of Spent Fuel Pool Cooling PRA: Model and Results, INEL-96/0334, Idaho National Engineering Laboratory, September 1996.

[2] N Siu et al, op cit.

[3] P G Prassinos et al, Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants, NUREG/CR-5176, January 1989.

this pool, the NRC sought to obtain knowledge that would be relevant to other PWRs.

Earthquake

The NRC's analysis of the Robinson pool showed that there is high confidence (95 percent) of a low probability (5 percent) of structural failure of the pool in the event of an earthquake of 0.65g. A more severe earthquake could cause structural failure and water loss, and the mean probability of such an event was estimated to be 1.8×10^{-6} per reactor-year.

Cask drop

The NRC's analysts examined a four-foot drop of a 68-ton fuel shipping cask onto the wall of the Robinson fuel pool. They estimated that the wall would suffer significant damage. Cracking of the concrete, yield of reinforcing steel, and tearing of the liner could be expected. Loss of pool water could follow. The probability of this cask drop was not estimated.

Relevance of these findings to Harris

Each nuclear plant has specific design features. Thus, the findings from Robinson cannot be applied uncritically to Harris. Nevertheless, the Robinson findings suggest that the Harris fuel pools may be vulnerable to water loss in the event of a severe earthquake or a cask drop.

The Harris pools are partly below the site's grade level, and the tops of the fuel racks are at grade level. However, there are rooms and passages below the pools. Also, there are three deep cavities adjacent to the fuel handling building, where the containments for Units 2-4 were to have been constructed. Thus, the pools could drain below the tops of the fuel racks, partially or completely, if damaged by an earthquake or cask drop.

Administrative and technical measures are employed at Harris to prevent a cask drop onto a pool wall or into a pool. There is some probability that these measures will fail and a cask drop will occur. No PRA estimate of this probability is available. An NRC-sponsored analysis found the probability of structural failure from a cask drop at the Millstone and Ginna plants, prior to improvements, to be 3×10^{-5} per reactor-year.[4] After improvements, the

[4] V L Sailor et al, Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82, NUREG/CR-4982, July 1987, Table 2.10.

probability was estimated to be lower than 2×10^{-8} per reactor-year. Such a low probability is beyond the range of credibility of PRA techniques.

5. A pool accident induced by a reactor accident

The Harris IPE predicts a core damage frequency of 7×10^{-5} per reactor-year. It further predicts that 15 percent of core damage sequences would be accompanied by a significant degree of containment failure or bypass, with a total probability of about 1×10^{-5} per reactor-year.[5] The resulting releases could initiate a pool accident by precluding personnel access.

Radiation levels close to the plant

Figure C-1 shows the estimated whole-body dose to exposed persons following a severe reactor accident.[6] The dose shown is averaged over a range of meteorological conditions and a set of potential atmospheric releases (PWR 1-5) from the NRC's 1975 Reactor Safety Study. Those releases involved a cesium release fraction ranging from 1-50 percent. A similar figure could be drawn for the releases predicted by the Harris IPE, with a qualitatively similar result.

From Figure C-1 it will be seen that an unprotected person one mile from the plant will receive a whole-body dose of about 1,000 rem over one day. Closer to the plant, the dose will be much higher, as shown in Figure C-2.[7] It has been estimated that the dose rate within a reactor containment, following a severe accident, will be 4 million rem per hour.[8] Given containment failure or bypass, doses approaching this level could be experienced outside the containment, in locations such as the fuel handling building.

Health effects of high dose levels

A radiation dose of 500-1,000 rem will normally kill an adult person within a few weeks, due to bone marrow damage. Doses of 1,000-5,000 rem will damage the gastro-intestinal tract, causing extensive internal bleeding and

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[5] Release categories involving significant containment failure or bypass are, in descending order of estimated probability, RC-4, RC-5, RC-6, RC-1B, RC-4C and RC-3. Each of these categories involves a 100 percent release of noble gases. The CsI release fraction ranges from .001 percent (RC-6) to 59 percent (RC-5).
[6] Figure C-1 is adapted from Figure 3.5-10 of: B Shleien, Preparedness and Response in Radiation Accidents, US Department of Health and Human Services, August 1983.
[7] Figure C-2 is adapted from Slide 16 of: J A Martin et al, Pilot Program: NRC Severe Reactor Accident Incident Response Training Manual, NUREG-1210, February 1987, Volume 4.
[8] R P Burke et al, In-Plant Considerations for Optimal Offsite Response to Reactor Accidents, NUREG/CR-2925, November 1982, Table B.2.

death within a few days. Doses above 10,000 rem will lead to failure of the central nervous system, causing death within a day.[9]

Prevention of access, and its implications

It is clear that a severe accident at the Harris reactor, accompanied by containment failure or bypass, would preclude personnel access to the plant. To this author's knowledge, CP&L has made no preparations to maintain pool cooling after such an event. It can be assumed that pool cooling would cease during the accident, and would not resume.

In CP&L's application for a license amendment to activate pools C and D at Harris, the bounding decay heat load for pools C and D is estimated to be 15.6 million BTU/hour (4.6 MW). CP&L states that the mass of water in these two pools, above the racks, will be 2.9 million pounds (1,320 tonnes). Then, CP&L estimates that the pools will begin to boil, if pool cooling systems become inoperative, after a period "in excess of 13 hours".[10] If we assume that cooling remains inoperative, and that 4.6 MW of heat is solely devoted to boiling off 1,320 tonnes of water, then this water will be entirely evaporated over a period of 180 hours (7.5 days). In practice, a slightly longer period will be required, accounting for heat losses.

Thus, a severe reactor accident with containment failure or bypass would lead to uncovering of spent fuel in the Harris pools, after a time delay of perhaps 10 days. Heroic efforts would be needed to restore cooling or to replace evaporated water. If these efforts involved addition of water to the pools after the fuel had been uncovered, they would run the risk of exacerbating the accident by inhibiting convective circulation of air in the pools (see Appendix D).

6. A sabotage/terrorism event involving siphoning

Appendix B discusses the potential for acts of malice at nuclear plants. A potential act of this kind at Harris would involve a group taking control of the fuel handling building, shutting down the pool cooling systems, and siphoning water from the pools. The consequent uncovering of fuel could initiate an exothermic reaction in recently discharged fuel within a few hours (see Appendix D). Once such a reaction was initiated, access to the fuel handling building would be precluded. Over the subsequent hours, exothermic reactions would be initiated in older fuel.

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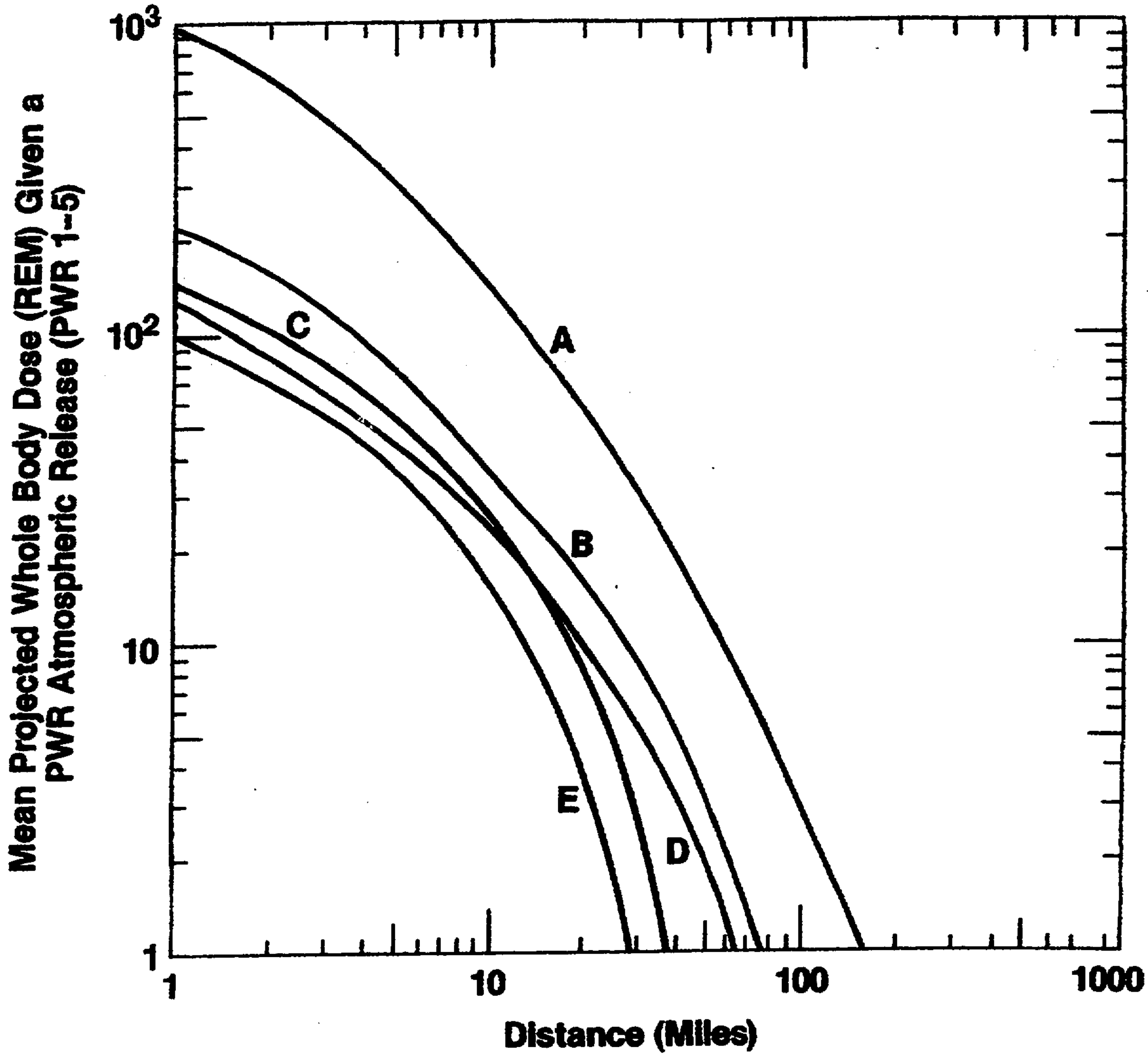
[9] B Flowers et al, Royal Commission on Environmental Pollution, Sixth Report, Cmnd. 6618, Her Majesty's Stationery Office, London, September 1976, page 23.
[10] License amendment application, Enclosure 7, page 5-8.

The group would require military skills and equipment to take control of the fuel handling building. Siphoning water from the pools would be a comparatively easy task. Escape by the group would be difficult but not impossible. The probability of this scenario cannot be predicted by PRA techniques.



Curve A	Individual located outdoors without protection. SF's (1.0, 0.7). 1-day exposure to radionuclides on ground.
Curve B	Sheltering, SF's (0.75, 0.33), 6-hour exposure to radionuclides on ground.
Curve C	Evacuation, 5-hour delay time, 10 mph.
Curve D	Sheltering, SF's (0.5, 0.08), 6-hour exposure to radionuclides on ground.
Curve E	Evacuation, 3-hour delay time, 10 mph.

Figure C-1

Estimated whole-body dose after a severe PWR accident

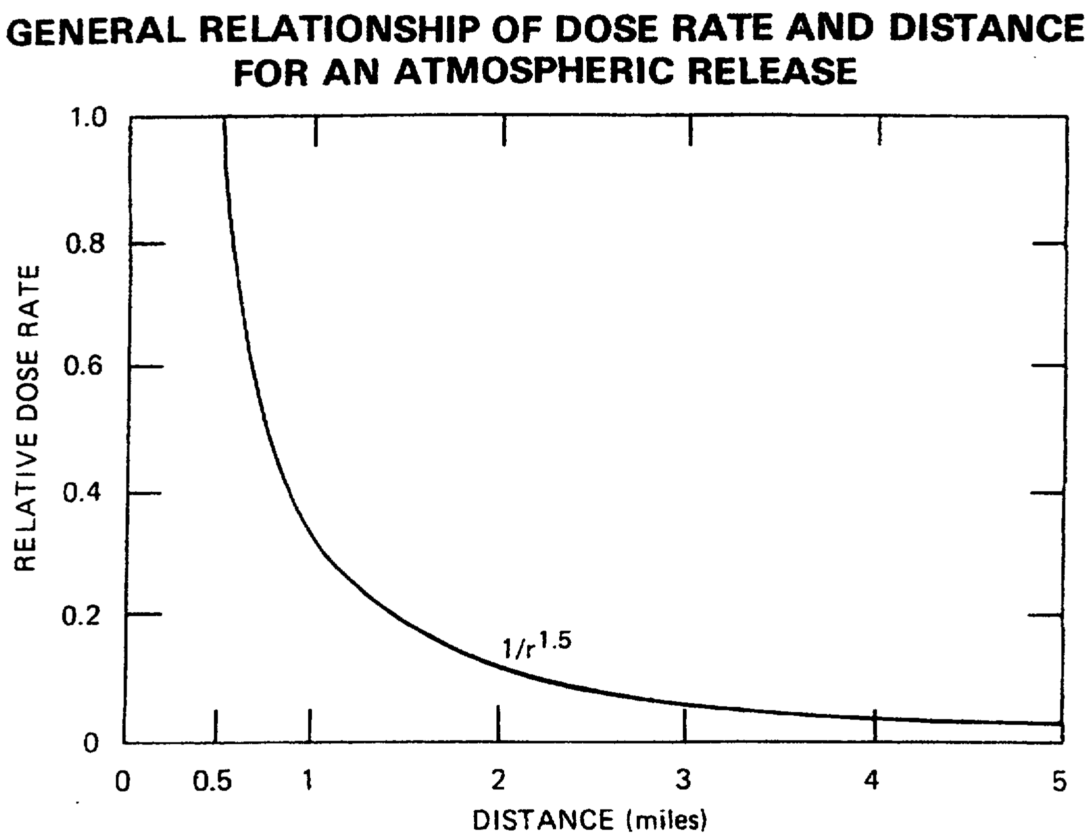


Figure C-2

Dose-distance relationship for a severe reactor accident

RISKS AND ALTERNATIVE OPTIONS
ASSOCIATED WITH SPENT FUEL STORAGE AT THE
SHEARON HARRIS NUCLEAR POWER PLANT

Appendix D

Potential for exothermic reactions in the Harris pools

1. Introduction

If water is totally or partially lost from one or more of the Harris fuel pools, the potential exists for an exothermic reaction between the fuel cladding and air or steam. The cladding is a zirconium alloy that begins to react vigorously with air or steam when its temperature reaches 900-1,000 degrees C. Partial or total loss of water could cause the cladding to reach this temperature, because water is no longer available to remove decay heat from the fuel. If the cladding temperature reaches 900-1,000 degrees C and air or steam remain available, a runaway reaction can occur. Heat from the exothermic reaction can increase cladding temperature, which will in turn increase the reaction rate, resulting in a runaway reaction.

The steam-zirconium reaction will be familiar to many observers of the 1979 TMI accident. During that accident a steam-zirconium reaction contributed to the partial melting of the reactor core, and generated hydrogen gas. Accumulation of this gas in the upper part of the reactor pressure vessel was a cause of concern during the accident. Hydrogen entered the containment and exploded about 10 hours into the accident, yielding a pressure spike of 28 psig.[1]

The potential for a partial or total loss of water from the Harris pools is addressed in Appendix C. Here, the consequent potential for exothermic reactions is considered. Also, this appendix considers the potential for exothermic reactions to release radioactive material -- especially the radioisotope cesium-137 -- from spent fuel to the atmosphere outside the Harris plant.

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[1] G Thompson, Regulatory Response to the Potential for Reactor Accidents: The Example of Boiling-Water Reactors, Institute for Resource and Security Studies, Cambridge, MA, February 1991.

2. Configuration of the Harris pools

A plan view of the Harris fuel handling building is provided in Figure A-1 of Appendix A. Figure D-1 shows a typical rack used in the Harris fuel pools. Carolina Power & Light Company (CP&L) has not published detailed information about the dimensions and configuration of the Harris racks, claiming that this information is proprietary. The center-center distances in the Harris racks are described in Appendix A.

Figure D-2 shows CP&L's intentions regarding placement of racks in pool C at Harris. It will be noted that the largest gap between the racks and the pool wall will be 2.4 inches, while the gap between racks will typically be 0.6 inches. In other words, the pool will be tightly packed with racks. Moreover, the racks will be tightly packed with fuel.

Effect of pool configuration on convective heat transfer

Examination of Figures D-1 and D-2 shows that convective circulation of air or water through the racks is limited to one pathway. Water (if the pool is full) or air (if the pool is empty) must enter the racks from below and pass upward through the fuel spaces. During Phases I and II of rack placement in pool C, air or water could reach the base of the racks from parts of the pool without racks. After racks are placed in Phase III, air or water must pass downward in the gap (1.4-2.4 inches) between the racks and the pool wall, and then travel horizontally across the bottom of the pool before entering racks from below.

It is further evident that the presence of residual water in the lower part of the pool would prevent convective circulation of air through the racks, in any of the three phases of rack placement. In this case, the only significant source of convective cooling would be from steam rising through the racks. This steam would be generated by the passage of heat from fuel assemblies to residual water, via conduction or thermal radiation.

Heat transfer pathways

Heat will be generated in the fuel assemblies by radioactive decay. Also, heat will be generated by exothermic reactions with zirconium, if these reactions are initiated. In the event of partial or total loss of water from a pool, the following pathways will be available to remove heat from the fuel assemblies, assuming that the assemblies remain intact:

(a) upward convection of air (for total loss of water) or steam (for partial loss of water);
(b) upward or downward conduction along the fuel rods and rack structure;
(c) upward or downward thermal radiation along the narrow passages between fuel rods, and between assemblies and rack walls;
(d) upward thermal radiation from the top of the racks to the interior of the fuel handling building;
(e) downward thermal radiation from the bottom of the racks to the base of the pool or to residual water (if present); and
(f) lateral conduction and thermal radiation across the racks to the pool wall.

For a fuel assembly separated from the pool wall by more than a few spaces, pathway (f) will be ineffective. Thus, only pathways (a) through (e) need to be considered. In the event of total loss of water, the effectiveness of pathway (a) will depend upon the extent of ventilation in the fuel handling building.

3. A scoping approach to heat transfer

To assess the effectiveness of the above-mentioned heat transfer pathways, it is appropriate to begin with a scoping analysis. Detailed calculations, especially if they involve computer modelling, must be guided by physical insight. Scoping calculations can help to provide that insight.

Decay heat output

The first parameter to be considered -- designated here as Q -- is the decay heat in a spent fuel assembly. The unit of Q is kW per metric ton of heavy metal (MTHM) in the assembly. For PWR fuel, Q is about 10 kW/MTHM for fuel aged 1 year from discharge, and about 1 kW/MTHM for fuel aged 10 years.[2]

Upper bound of temperature rise

Now consider a fuel pellet which is in complete thermal isolation. Due to decay heat, this pellet will experience a temperature rise of 11Q degrees C per hour.[3] Thus, if Q=10, the temperature rise will be 110 degrees C per hour (2,640 degrees C per day). A temperature rise of 11Q degrees C per hour is the

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[2] For fuel burnups typical of current practice, Q will actually be 10-20 percent higher than the values shown here.
[3] Assuming that a uranium dioxide pellet has a specific heat of 300 J/K per kg of pellet (340 J/K per kg of HM).

upper bound to the temperature rise that could be experienced by a fuel assembly, absent the initiation of an exothermic reaction of the cladding.

Heat transfer by conduction

Next, consider conduction along the fuel rods. A Harris PWR assembly has 264 rods, each containing 1.74 kg of HM. Each rod is 12 ft long, with an outer diameter of 0.374 inches, a cladding thickness of 0.0225 inches, and a pellet diameter of 0.3225 inches.[4] Assume that decay heat is generated uniformly along the length of the rod, conduction along the rod is the only heat transfer mechanism, and the two ends of the rod have the same temperature, Y (degrees C). Then, the temperature at the middle of the rod will be Y+2,000Q degrees C.[5] This result could be viewed as counter-intuitive, because the decay heat in each rod is only 0.48Q Watts per meter of rod.

Convective cooling by steam

Now consider convective cooling of a fuel assembly by upward motion of steam that is generated from residual water at the lower end of the assembly. Neglect other heat transfer mechanisms, assume that decay heat is generated uniformly along the length of the fuel rods, and assume that the temperature of the residual water is 100 degrees C. Define S as the submerged fraction of the assembly and T (degrees C) as the temperature of steam leaving the top of the fuel assembly. Neglect the thermal inertia of the pellets and cladding. Then, the amount of steam generated is proportional to S, while the decay heat captured by this steam is proportional to (1-S). It follows that:[6]

$$T = 100 + (2{,}260/2.1) \times [(1-S)/S]$$

Note that Q does not enter this equation. If one-tenth of a fuel assembly is submerged (S = 0.1), this equation yields a T of 9,800 degrees C. A temperature of this magnitude would not be generated in practice, because of thermal inertia and the operation of other heat transfer mechanisms.[7] However, the calculation establishes an important point. Convective cooling of fuel assemblies by steam from residual water will be ineffective when the submerged fraction of the assemblies is small.

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4 Harris FSAR, Section 1.3, Amendment No. 30.

5 Assuming that the cladding's thermal conductivity is 17.3 W/mK, the pellets' conductivity is 1.99 W/mK, and pellets are in perfect contact with each other and the cladding.

6 Assuming that the latent heat of evaporation of water is 2,260 kJ/kg and the specific heat of steam is 2.1 kJ/kgK.

7 The singularity of the T equation at S=0 reflects the lack of consideration of other heat transfer mechanisms.

Cooling by thermal radiation

If residual water is present, there remains only one potentially effective mechanism of heat transfer from the mid-length of a fuel assembly -- thermal radiation along the axis of the assembly. Note that a Harris PWR assembly has an active length of 12 feet, a cross-section 8.4 inches square, and contains 264 fuel rods plus other longitudinal structures. In the Harris fuel pools, the assembly will be surrounded by continuous sheets of neutron-absorbing material (Boral), and the center-center distance in pool C will be 9.0 inches. In this configuration, axial heat transfer by thermal radiation will be strongly inhibited. However, calculations more detailed than those above are required to estimate the amount of heat that can be transferred by this pathway.

Note that downward heat transfer by radiation will increase the generation of steam from residual water, thus improving the effectiveness of convective cooling by steam. A detailed analysis should consider such effects through coupled calculations.

Summary

The preceding scoping calculations show that conduction and convective cooling by steam will be relatively ineffective. These cooling mechanisms cannot prevent fuel cladding from reaching a temperature of at least 1,000 degrees C -- the initiation point for a runaway exothermic reaction -- even for fuel aged in excess of 10 years. An estimate of the effectiveness of axial radiation cooling -- the only remaining cooling mechanism if residual water is present -- would require more detailed calculations. However, this author does not expect that such calculations would show axial radiation cooling to be more effective than conduction or convective cooling by steam.

If residual water is not present, a fuel assembly can be cooled by convective circulation of air. Estimation of the effectiveness of this mechanism requires an analysis of convective circulation through the pool and the fuel handling building, reflecting practical factors such as constrictions at the base of fuel racks.

4. Specifications for an adequate, practical analysis

There has been no site-specific analysis of the potential for exothermic reactions in the Harris pools. Generic analyses have been performed for and by the US Nuclear Regulatory Commission (NRC). Before addressing the findings and adequacy of the NRC's generic analyses, let us consider the ingredients that are necessary if an analysis is to provide practical guidance about the potential for exothermic reactions in the Harris spent fuel pools. Sections 2 and 3 of this appendix provide a basis for specifying those ingredients.

Partial and complete uncovering of fuel

First, the analysis should not be limited to instantaneous, complete loss of water from a pool. Such a condition is unrealistic in any accident scenario which preserves the configuration of the spent fuel racks. If water is lost by drainage or evaporation and no makeup occurs, then complete loss of water will always be preceded by partial uncovering of the fuel. If makeup is considered, the water level could fall, rise or remain static for long periods.

Partial uncovering of the fuel will often be a more severe condition than complete loss of water. As shown above, convective heat loss is suppressed by residual water at the base of the fuel assemblies. As a result, longer-discharged fuel with a lower Q may undergo a runaway steam-zirconium reaction during partial uncovering while it would not undergo a runaway air-zirconium reaction if the pool were instantaneously emptied.

In a situation of falling water level, a fuel assembly might first undergo a runaway steam-zirconium reaction, then switch to an air-zirconium reaction as water falls below the base of the rack and convective air flow is established. In this manner, a runaway air-zirconium reaction could occur in a fuel assembly that is too long-discharged (and therefore has too low a Q) to suffer such a reaction in the event of instantaneous, complete loss of water. Conversely, a rising water level could precipitate a runaway steam-zirconium reaction in a fuel assembly that had previously been completely uncovered but had not necessarily suffered a runaway air-zirconium reaction while in that condition. The latter point is highly significant in the context of emergency measures to recover control of a pool which has experienced water loss. Inappropriate addition of water to a pool could exacerbate the accident.

Computer modelling

An adequate analysis of the potential for exothermic reactions will require computer modelling. The modelling should consider both partial and complete uncovering and the transition from one of these states to the other. Also, the modelling should cover: (a) thermal radiation, conduction, and steam or air convection; (b) air-zirconium and steam-zirconium reactions; (c) variations along the fuel rod axis; (d) radial variations within a representative fuel rod, including effects of the pellet-cladding gap; and (e) clad swelling and rupture. Experiments will probably be required to support and validate the modelling.

Site-specific factors

The analysis can be strongly influenced by site-specific factors. For convective cooling by air, these factors include the detailed configuration of the racks, the pools and the fuel handling building. All relevant factors should be accounted for. This could be done through site-specific modelling. Alternatively, generic modelling could be performed across the envelope of site-specific parameters, with sensitivity analyses to show the effects of varying those parameters.

Propagation of exothermic reactions to adjacent assemblies

After an exothermic reaction has been initiated in a group of fuel assemblies, this reaction might propagate to adjacent assemblies. Due to their lower Q or to other factors, the adjacent assemblies might not otherwise suffer an exothermic reaction. An analysis of propagation should consider the potential for reactions involving not only the fuel cladding but also material (e.g., Boral) in the fuel racks. The analysis should examine the implications of clad and pellet relocation after a reacting assembly has lost its structural integrity. Those implications include the heating of adjacent assemblies and racks by direct contact, thermal radiation, convection, and the inhibition of air circulation. A bed of relocated material at the base of the pool could have all these effects.

5. The 1979 Sandia study

An initial analysis of the potential for exothermic reactions was made for the NRC by Sandia Laboratories in 1979.[8] This was a respectable analysis as a first attempt. It considered partial drainage of a pool, although it used a crude heat transfer model to study that problem, and neglected to consider the steam-zirconium reaction. It did not address the potential for propagation of exothermic reactions to adjacent assemblies. The Sandia authors were careful to state their assumptions and to specify the technical basis for their computer modelling.

Figure D-3 illustrates the findings of the Sandia study. The three lower curves in Figure D-3 show the sensitivity of convective air cooling to the diameter of the hole in the base of the fuel racks. The next higher curve -- the

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[8] A S Benjamin et al, Spent Fuel Heatup Following Loss of Water During Storage, NUREG/CR-0649, March 1979.

"blocked inlets" case -- shows the suppression of convective air cooling due to the presence of residual water. The dashed curve shows the effect of an air-zirconium reaction. The runaway nature of that reaction is evident.

Note that the analysis underlying Figure D-3 assumed a cylindrical rack arrangement with a center-center distance of about 13 inches. Also, the analysis assumed a gap of 16 inches between the racks and the pool wall. The Harris racks are more compact and are packed more tightly into their pools. These factors will tend to inhibit convective air cooling at Harris.

6. Subsequent studies

The 1979 Sandia study could have been the first of a series of studies that moved toward the level of adequacy specified in Section 4. Since 1979 the NRC has sponsored or performed a variety of studies related to the initiation of exothermic reactions in fuel pools.[9] However, the scope of these studies has narrowed, and their potential for building on the 1979 study has not been realized.

Failure to consider partial uncovering

A major weakness of the NRC's studies since 1979 has been their focus on a postulated scenario of total, instantaneous loss of water. This appendix shows clearly that partial uncovering of fuel will often be a more severe condition than complete loss of water. Thus, however sophisticated the NRC's modelling of spent fuel heatup might be, the findings have limited relevance to the practical potential for exothermic reactions.

Brookhaven National Laboratory (BNL) has developed the SHARP code to replace the SFUEL code first developed at Sandia. BNL authors have claimed that the SHARP code can more accurately predict spent fuel heatup in realistic spent fuel pool configurations.[10] A review of the SHARP code is beyond the scope of this report. Applied to spent fuel in a generic, high-density configuration in an instantaneously emptied pool, the SHARP code finds that the fuel cladding will reach a "critical" temperature (565 degrees C) if aged less than 17 months for PWR fuel or 7 months for BWR fuel.[11] The relevance of this finding to the Harris pools is unclear.

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[9] See, for example: V L Sailor et al, Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82, NUREG/CR-4982, July 1987; and R J Travis et al, A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants, NUREG/CR-6451, August 1997.
[10] R J Travis et al, page 3-4.
[11] Ibid.

Propagation of exothermic reactions

Pursuant to a Freedom of Information request, the NRC released in 1984 a so-called draft report by MIT and Sandia authors on the propagation of an air-zirconium reaction in a fuel pool.[12] This document has been repeatedly cited in subsequent years, although it should properly be regarded as notes toward a draft report. Those notes were submitted to the NRC after the project ran out of funds; it was never completed.

The MIT-Sandia group concluded from computer modelling and experiments that an air-zirconium reaction in fuel assemblies could propagate to adjacent, lower-Q assemblies. They expressed the view that propagation would be quenched in regions of a pool where fuel is aged 3 years or more, but noted the presence of "large uncertainties" in their analysis.

BNL analysts subsequently reviewed these experiments and conducted their own modelling using the same code (SFUEL). In their modelling the BNL analysts chose to terminate the air-zirconium reaction when the cladding reached its melting point.[13] Neither the MIT-Sandia group nor the BNL group examined the implications of clad and pellet relocation after a reacting assembly has lost its structural integrity. The author is not aware of other analyses which address this problem. Thus, the specifications set forth in Section 4 for analysis of propagation have not been met.

7. The potential for an atmospheric release of radioactive material

Spent fuel at Harris which suffers an exothermic reaction will release radioactive material to the fuel handling building. That building is not designed as a containment structure, and is not likely to be effective in this role, given the occurrence of exothermic reactions in one or more pools. A BNL study has concluded that a reasonable, generic estimate of the release fraction of cesium isotopes, from affected fuel to the atmosphere outside the plant, is 100 percent.[14] This release fraction is used in Appendix E.

The amount of fuel that will suffer an exothermic reaction, given a loss of water from the Harris pools, will depend upon the particular scenario. For scenarios which involve partial uncovering of fuel, the reaction could affect fuel aged 10 or more years. For scenarios which involve total loss of water,

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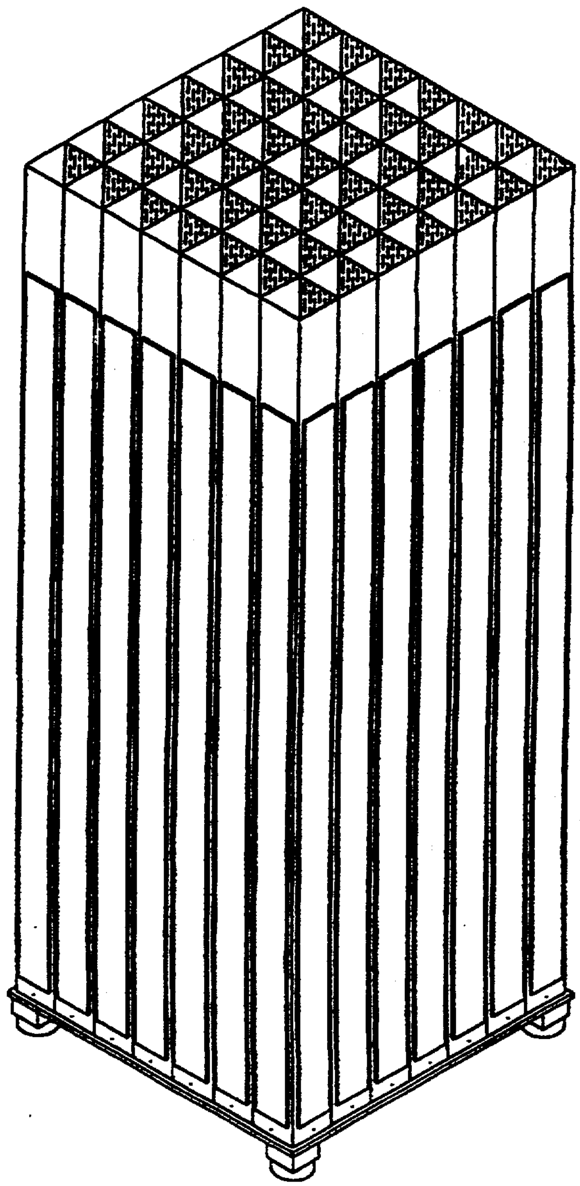
[12] N A Pisano et al, <u>The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool</u>, Draft Report, January 1984.
[13] V L Sailor et al.
[14] Ibid.

the reaction will be initiated only in younger fuel, perhaps aged no more than 1-2 years. However, if clad/pellet relocation is properly factored into a propagation analysis, this analysis may show that a reaction will propagate to much older fuel.

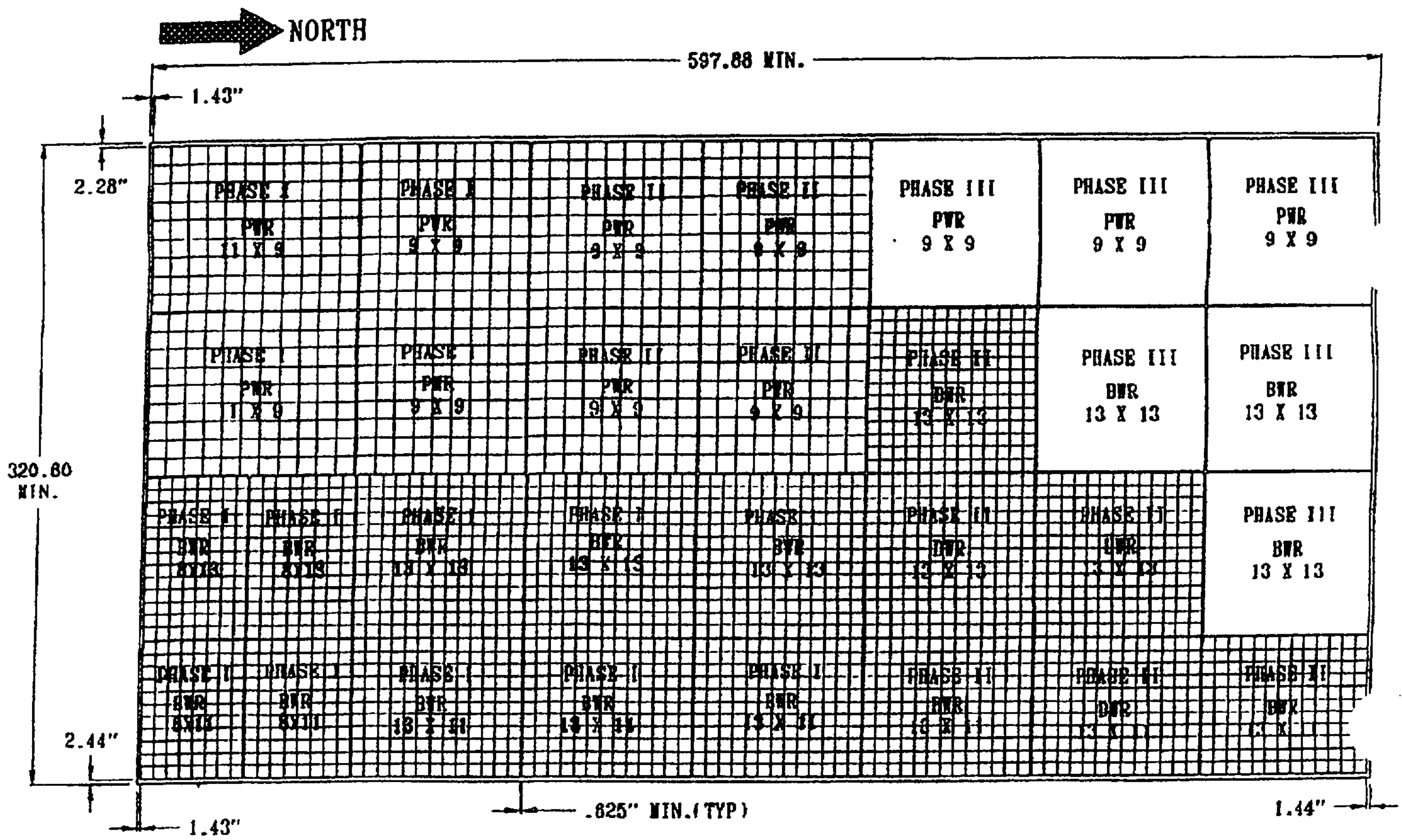
Appendix E considers two potential releases of cesium-137 from the Harris pools. One release corresponds to an exothermic reaction in fuel aged 9 years or less. The other release corresponds to a reaction in fuel aged 3 years or less.



Source: License amendment application

Figure D-1

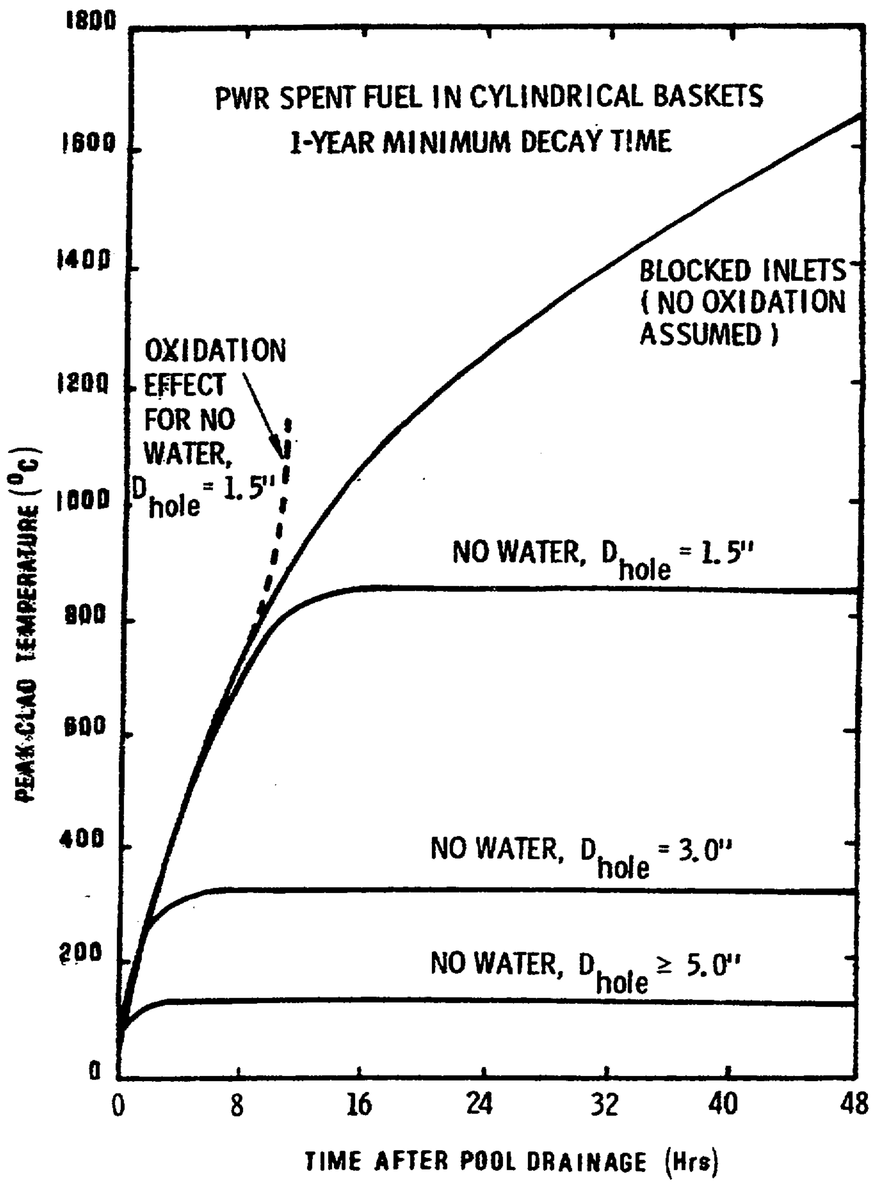
Typical rack used in the Harris pools



Source: License amendment application

Figure D-2

Proposed rack placement in Harris pool C



Source: NUREG/CR-0649

Figure D-3

Estimated heatup of PWR spent fuel after water loss

RISKS AND ALTERNATIVE OPTIONS
ASSOCIATED WITH SPENT FUEL STORAGE AT THE
SHEARON HARRIS NUCLEAR POWER PLANT

Appendix E

Consequences of a large release of cesium-137 from Harris

1. Introduction

This appendix outlines some of the potential consequences of postulated large releases of cesium-137 from the Harris plant to the atmosphere. Such consequences can be estimated by site-specific computer models. A simpler approach is used here, but this approach is adequate to show the nature and scale of expected consequences.

2. Characteristics of postulated releases

Two spent fuel release scenarios are postulated here. The first scenario involves a release of 2.3×10^7 Curies (870,000 TBq) of cesium-137, with a mass of 260 kilograms.[1] This represents the cesium-137 inventory in Harris' stock of spent fuel aged 3 years or less, as estimated in Appendix A. The second scenario involves a release of 7.1×10^7 Curies (2,600,000 TBq) of cesium-137, with a mass of 790 kilograms. This represents the cesium-137 inventory in Harris' stock of spent fuel aged 9 years or less. Note that all of the cesium-137 in the affected fuel is assumed to reach the atmosphere, an assumption which is explained in Appendix D.

Releases of the postulated magnitude could occur as a result of exothermic reactions in the Harris fuel pools. Appendix D discusses the potential for such reactions. Cesium-137 would not be the only radioisotope released to the atmosphere if exothermic reactions occurred in the pools. However, cesium-137 is likely to be the dominant cause of offsite radiological exposure,

[1] 1 Curie is equivalent to 3.7×10^{-2} TBq. 1 TBq of cesium-137 is equivalent to 0.3 grams.

just as it dominates the offsite exposure attributable to the 1986 Chernobyl reactor accident.[2] Note that cesium-137 has a half-life of 30 years.

A severe accident at the Harris reactor could also release cesium-137 to the atmosphere. Appendix A notes that the US Nuclear Regulatory Commission (NRC) has estimated the inventory of cesium-137 in the core of the Harris reactor, during normal operation, to be to be 4.2×10^6 Curies (155,000 TBq, or 47 kilograms). As summarized in Appendix B, an individual plant examination (IPE) study by Carolina Power & Light Company (CP&L) has identified six categories of potential significant release due to severe accidents at the Harris reactor. Release category RC-5, the most severe release category, would involve a release to the atmosphere of 53-59 percent of the cesium isotopes in the reactor core. Thus, given the NRC's estimate of core inventory, release category RC-5 would involve an atmospheric release of 2.2-2.5×10^6 Curies (82,000-92,000 TBq, or 25-28 kilograms) of cesium-137.

Chernobyl and weapons testing releases

For comparison with the above-mentioned potential releases, consider two actual releases -- from the Chernobyl accident and from atmospheric testing of nuclear weapons. The 1986 Chernobyl reactor accident released about 90,000 TBq (27 kilograms) of cesium-137 to the atmosphere, representing 40 percent of the cesium-137 in the reactor core.[3] Through 1980, about 740,000 TBq (220 kilograms) of cesium-137 were deposited as fallout in the Northern Hemisphere, as a result of atmospheric testing of nuclear weapons.[4] Note that the fallout from weapons testing was distributed over a larger area than the fallout from the Chernobyl accident, and a larger fraction of it descended on oceans and lightly inhabited areas.

3. Contamination of land

A useful indicator of the consequences of a cesium-137 release is the area of contaminated land. Here, contamination is measured by the external (whole-body) radiation dose that people will receive if they live in a contaminated area. When cesium-137 is deposited from an airborne plume, it will adhere to the ground, vegetation and structures. From these locations, it will emit gamma radiation which provides an external radiation dose to an exposed person. Cesium-137 will also enter the food chain and water sources, thereby

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[2] US Department of Energy, Health & Environmental Consequences of the Chernobyl Nuclear Power Plant Accident, DOE/ER-0332, June 1987; A S Krass, Consequences of the Chernobyl Accident, Institute for Resource & Security Studies, Cambridge, MA, December 1991.
[3] Krass, op cit.
[4] US Department of Energy, op cit.

providing an internal radiation dose to a person living in the contaminated area. Absent any countermeasures, the internal dose could be of a similar magnitude to the external dose.

Figure E-1 shows the relationship between contaminated land area and the size of an atmospheric release of cesium-137. This figure is adapted from a 1979 study by Jan Beyea, then of Princeton University.[5] The threshold of contamination is an external dose of 10 rem over 30 years, assuming a shielding factor of 0.25 and accounting for weathering of cesium. The "typical meteorology" case in Figure E-1 assumes a wind speed of 5 m/sec, atmospheric stability in class D, a 0.01 m/sec deposition velocity, a 1,000 m mixing layer and an initial plume rise of 300 m (although the results are not sensitive to plume rise). A Gaussian, straight-line plume model was used, providing an estimate of contaminated land area that will approximate the area contaminated during a range of actual meteorological conditions. The lower and upper limits of land contamination in Figure E-1 represent a range of potential meteorological conditions.

The threshold for land contamination

An external exposure of 10 rem over 30 years would represent about a three-fold increase above the typical level of background radiation (which is about 0.1 rem/year). In its 1975 Reactor Safety Study, the NRC used a threshold of 10 rem over 30 years as an exposure level above which populations were assumed to be relocated from rural areas. The same study used a threshold of 25 rem over 30 years as a criterion for relocating people from urban areas, to reflect the assumed greater expense of relocating urban inhabitants.

In an actual case of land contamination in the United States, the steps taken to relocate populations and pursue other countermeasures (decontamination of surfaces, interdiction of food supplies, etc.) would reflect a variety of political, economic, cultural, legal and scientific influences. It is safe to say that few citizens would calmly accept a level of radiation exposure which substantially exceeds background levels.

Land contamination from potential Harris releases

Three potential Harris releases of cesium-137 are shown in Figure E-1. Releases of 70 million Curies and 20 million Curies correspond to liberation

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[5] J Beyea, "The Effects of Releases to the Atmosphere of Radioactivity from Hypothetical Large-Scale Accidents at the Proposed Gorleben Waste Treatment Facility", in Chapter 3 of Report of the Gorleben International Review, presented (in German) to the Government of Lower Saxony, March 1979.

of cesium-137 from spent fuel aged up to 9 years or up to 3 years, respectively. A release of 2 million Curies corresponds to the most severe reactor accident identified in the Harris IPE.

For typical meteorology, Figure E-1 indicates that a release of 2 million Curies would contaminate 4,000-5,000 square kilometers of land, A release of 20 million Curies would contaminate 50,000-60,000 square kilometers. Finally, a release of 70 million Curies would contaminate about 150,000 square kilometers of land. Note that the total area of North Carolina is 136,000 square kilometers and the state's land area is 127,000 square kilometers.[6]

Potentially exposed population

According to CP&L's Final Safety Analysis Report (FSAR) for the Harris plant, an estimated 1.8 million people will live within 50 miles of the plant in 2000, while 2.2 million people will live within that radius in 2020.[7] A 50 mile-radius circle encompasses an area of 20,300 square kilometers.

If a substantial release of cesium-137 occurs at Harris, the shape and size of the resulting contaminated area will depend on the size of the release and the meteorological conditions during the period of the release. If the wind direction is constant during the release and the atmosphere remains stable, the contaminated area will be comparatively narrow and extended downwind. Changing wind direction during the release period and a less stable atmosphere will produce a more "smeared out" pattern of contamination.

A computer modelling exercise could be performed, to predict patterns of contamination under different meteorological conditions. This exercise could ascribe a probability, assuming a postulated release, that a particular population falls within an area contaminated above a specified threshold.

4. Health effects of radiation

The health effects of exposure to ionizing radiation can be broadly categorized as early and delayed effects. For our postulated releases of cesium-137, early health effects could be suffered by some people in the immediate vicinity of the plant. However, most of the health effects would be delayed effects, especially cancer, which are manifested years after the initial exposure.

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6 The World Almanac and Book of Facts 1991, Pharos Books, New York, 1990.
7 Harris FSAR, Section 2.1.3, Amendment No. 2.

Note that a release during a reactor accident (e.g., release category RC-5 at Harris) will contain short-lived radioisotopes as well as cesium-137. Under certain conditions of meteorology and emergency response, the presence of these short-lived radioisotopes in the release could cause many early health effects. Spent fuel contains comparatively small amounts of short-lived radioisotopes. Thus, early health effects are comparatively unlikely if a release occurs from a spent fuel pool.

Table E-1 shows an estimate of the excess cancer mortality attributable to continuous exposure to a relatively low radiation dose rate. This estimate was made by the BEIR V committee of the National Research Council.[8] In Table E-1, a continuous exposure of 1 mSv/year (0.1 rem/year) is assumed to occur throughout life.[9] Such an exposure is estimated to increase the number of fatal cancers, above the normally expected level, by 2.5 percent for males and 3.4 percent for females, with an average of 16-18 years of life lost per excess death. If the dose-response function were linear, it would follow that continuous, lifetime exposure to 10 mSv/year (1 rem/year) would increase the number of fatal cancers by 25 percent for males and 34 percent for females. The shape of the dose-response function is a subject of ongoing debate.

If people continued to occupy urban areas contaminated with cesium-137 to an external exposure level just below 25 rem over 30 years, as was assumed in the Reactor Safety Study, their average exposure during this 30-year period would be 8 mSv/year (0.8 rem/year). An additional, internal exposure would arise from contamination of food and water. After 30 years, rates of external and internal exposure would decline, consistent with the decay of cesium-137. Note that over a period of 300 years (10 half-lives), the activity of cesium-137 will decay to one-thousandth of its initial level.

5. Economic consequences of a release of radioactivity

Computer models have been developed for estimating the economic consequences of large atmospheric releases of radioactive materials. Findings from such models have been used by the NRC to evaluate the cost-benefit ratio of introducing measures to reduce the probabilities or consequences of spent fuel pool accidents.[10] A review of these models, findings and cost-

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[8] National Research Council, Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V, National Academy Press, Washington, DC, 1990. Table E-1 is adapted from Table 4-2 of the BEIR V report.

[9] The exposure of 1 mSv/year is additional to background radiation, whose effects are accounted for in the normal expectation of cancer mortality.

[10] See, for example: E D Throm, Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools, NUREG-1353, April 1989; and J H Jo et al,

benefit analyses is beyond the scope of this report. However, a brief examination of the NRC's literature reveals that findings in this area rest on assumptions and value judgements that are not clearly articulated and deserve thorough public review.

Previous sections of this appendix have shown that potential releases of cesium-137 from the Harris spent fuel pools could lead to the relocation of large populations and ongoing radiation exposure to large, unrelocated populations. Relocation implies abandonment of large amounts of land, other natural resources and fixed capital. Political and social effects would be significant, and would have economic implications. Useful analysis of these matters would require a more sophisticated approach than is evident in literature generated by and for the NRC.

Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools, NUREG/CR-5281, March 1989.

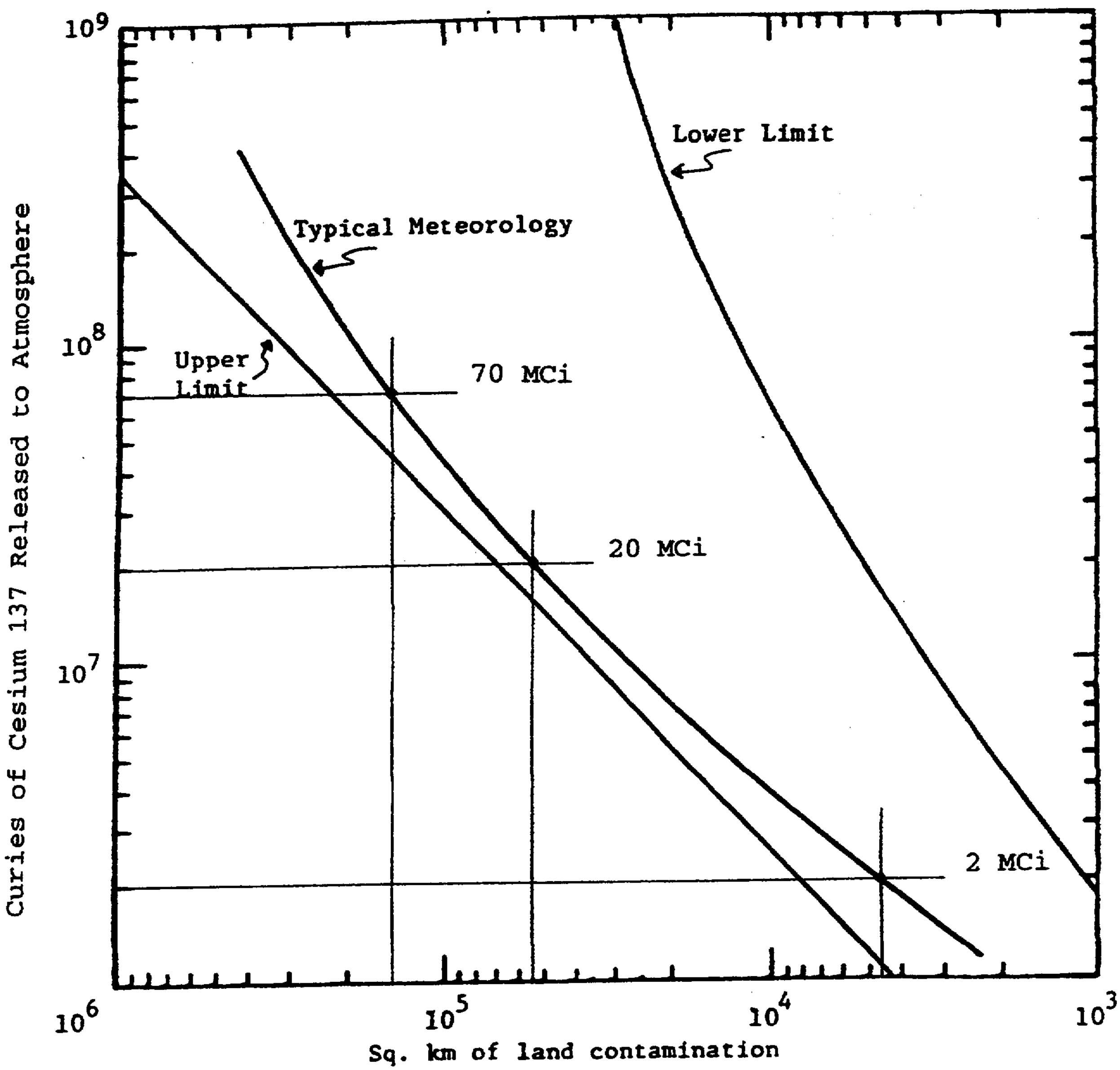

Figure E-1

Contaminated land area as a function of cesium-137 release

ESTIMATED LIFETIME RISK PER 100,000 PERSONS EXPOSED TO 1 mSv PER YEAR, CONTINUOUSLY THROUGHOUT LIFE

	Males	**Females**
• Point estimate of excess mortality	520	600
• 90 percent confidence limits	410-980	500-930
• Normal expectation	20,560	17,520
• Excess as percent of normal	2.5	3.4
• Average years of life lost per excess death	16	18

Table E-1

Excess cancer mortality from continuous exposure to radiation: BEIR V estimate

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISISON
Before the
ATOMIC SAFETY AND LICENSING BOARD

In the matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY	)	**Docket No. 50-400**
	)	
(Harris Nuclear Plant)	)	**March 31, 1999**

DECLARATION OF DAVID A. LOCHBAUM, NUCLEAR SAFETY ENGINEER, UNION OF CONCERNED SCIENTISTS, CONCERNING TECHNICAL ISSUES AND SAFETY MATTERS INVOLVED IN THE HARRIS NUCLEAR PLANT LICENSE AMENDENT FOR SPENT FUEL STORAGE

I, David A. Lochbaum, make the following declaration:

1. My name is David A. Lochbaum. I reside in the state of Maryland.

2. I am employed by the Union of Concerned Scientists as its nuclear safety engineer. I have been so employed since October 1996. I have the following responsibilities: a) direct and coordinate UCS's nuclear safety program; b) monitor developments in nuclear industry to assess and respond to impact; c) serve as technical authority and spokesperson on nuclear issues; and d) initiate legal action to correct safety problems.

3. The Union of Concerned Scientists, with offices located at 1616 P Street NW Suite 310, Washington, DC 20036, is an independent nonprofit organization dedicated to advancing responsible public policies in areas where technology plays a critical role.

4. I have worked in the field of nuclear engineering since June 1979. I am a graduate of the University of Tennessee with a bachelor of science in nuclear engineering.

5. After receiving my nuclear engineering degree, I went to work for the Georgia Power Company as a junior engineer at their Edwin I. Hatch Nuclear Power Plant. I held various positions in the commercial nuclear power industry over the next 17 years prior to joining UCS. This experience is detailed in the resume attached hereto as Exhibit A.

6. I am the author of *Nuclear Waste Disposal Crisis* (Pennwell Books, Tulsa, January 1996) on the technical problems with spent fuel storage at reactor sites.

7. I have examined the license amendment application dated December 23, 1998, submitted by the Carolina Power & Light Company (CP&L) to the Nuclear Regulatory Commission (NRC) concerning spent fuel storage at the Harris Nuclear Plant and the report prepared by Gordon Thompson of the Institute for Resource and Security Studies titled, "Risks and Alternative Options Associated with Spent Fuel Storage at the Shearon Harris Nuclear Power Plant," dated February 1999. I am familiar with these documents and have relied upon them in formulating the opinions contained in this declaration. I have also examined and am familiar with, for the purposes of preparing this declaration, the applicable federal regulations contained in Title 10 of the Code of Federal Regulations; NRC Information Notice No. 85-30: "Microbiologically Induced Corrosion of Containment Service Water System," dated April 19, 1985; NRC Information Notice No. 85-56: "Inadequate Environment Control For Components and Systems in Extended Storage or Layup," dated July 15, 1985; NRC Information Notice No. 94-38: "Results of

a Special NRC Inspection at Dresden Nuclear Power Station Unit 1 Following a Rupture of Service Water Inside Containment," dated May 27, 1994; NRC Inspection Manual Procedure 92050: "Review of Quality Assurance For Extended Construction Delay;" NRC Inspection Report No. 50-400/80-26, 50-401/80-24, 50-402/80-24, and 50-403/80-24, dated January 2, 1981; letter from Darrell G. Eisenhut, Nuclear Regulatory Commission, to Christopher John Adams, The Orange County Democratic Party, dated June 9, 1981; NRC Inspection Report No. 50-400/81-14, 50-401/81-14, 50-402/81-14, and 50-403/81-14, dated August 5 (or 25th), 1981; NRC Inspection Report No. 50-400/81-13, 50-401/81-13, 50-402/81-13, and 50-403/81-13, dated August 13, 1981; and NRC Inspection Report No. 50-400/81-15, 50-401/81-15, 50-402/81-15, and 50-403/81-15, dated September 14, 1981. I have also relied upon these documents in formulating my opinions as expressed in this declaration.

8. Having examined the relevant documents as mentioned above, it is my professional opinion that CP&L's proposed use of an alternative plan per 10 CFR 50.55a to demonstrate that the Unit 2 fuel pool cooling system was "designed, fabricated, erected, constructed and inspected to quality standards commensurate with the importance of the safety function to be performed"[1] raises significant safety concerns for persons living near the facility. It is also my professional opinion that these significant safety concerns have not been adequately considered in the license amendment application filed by CP&L. These concerns are set forth below. I recommend that they be

[1] 10 CFR 50.55a, Codes and standards.

included in the subject matter of issues to be considered by the Atomic Safety and Licensing Board in the above captioned proceeding.

9. It is my professional opinion that the Alternative Plan, as described in Enclosure 8 to the December 23, 1998, license amendment application submitted by CP&L, is deficient for the following reasons:

(a) CP&L notified the NRC in December 1981 that construction on Units 3 and 4 at the Harris Nuclear Plant had been cancelled. In December 1983, CP&L notified the NRC that Unit 2 had been cancelled. Unit 1 was completed and placed into commercial operation in May 1987.[2]

(b) The Alternative Plan describes the process CP&L proposes to certify the installed portions of the Unit 2 spent fuel pool cooling system at the Harris Nuclear Plant in lieu of the original construction records which purportedly were discarded in September 1993.[3]

(c) The Alternative Plan includes review of available documentation, inspection and/or examination of accessible components, internal (via camera) inspections of selected inaccessible components, and hydrostatic testing.

(d) The Alternate Plan and the license amendment application do <u>not</u> describe any program for proper storage and preservation of materials and components as required by Appendix B to 10 CFR Part 50. Nor do they describe any effort to determine if the installed piping and equipment experienced any deterioration over the many years of non-use since the piping and equipment were installed.

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[2] CP&L, Enclosure 8 to December 23, 1998 submittal, pp. 1-2.
[3] CP&L, Enclosure 8 to December 23, 1998 submittal, page 11.

(e) NRC Information Notice No. 85-30 documents a problem experienced at the H B Robinson nuclear plant, also owned and operated by CP&L, during 1984. According to this NRC document, stainless steel piping at the Robinson plant experienced significant corrosion pitting during an outage lasting about one year.

(f) The NRC issued Information Notice No. 85-56 "to alert addressees to problems which can occur if equipment is improperly stored or laid up during construction or extended plant outages. Addressees also are reminded that programs for proper storage and preservation of materials and components are required by NRC regulations (10 CFR 50, Appendix B), even though not specifically addressed as license conditions." According to this NRC document, a heat exchanger at Nine Mile Point Unit 2 was found to have experienced significant corrosion in the eight (8) years it had been stored in-place.

(g) NRC Information Notice No. 94-38 documents a problem at Dresden Unit 1. A water-filled pipe froze and ruptured, causing 55,000 gallons of water to flood the containment building. The NRC reported that the plant's owner had not taken adequate measures to protect the permanently closed facility. The NRC also reported that had a water-filled pipe connected to the spent fuel pool – which was as unprotected as the pipe which failed – ruptured, the water level in the spent fuel pool could have dropped below the top of its irradiated fuel assemblies.

(h) NRC Inspection Procedure 92050 contains guidance for NRC inspectors when auditing nuclear plants encountering extended construction delays. Among other areas, the guidance covers the plant owner's program for the protection and preservation of equipment. The elements of this program include:

1. Protective coverings and coatings,
2. Internal preservation,
3. Dunnage and other supports, and
4. Cleanliness preservation.

The Alternative Plan does not address any of these elements or describe measures taken to protect piping and components during the extended delay between construction and proposed use of this equipment.

10. NRC Inspection Report 50-400/80-26, 50-401/80-24, 50-402/80-24, and 50-403/80-24, dated January 2, 1981, transmitted a Notice of Violation to CP&L involving "failure to store equipment in accordance with instructions to prevent damage or deterioration."

11. NRC Inspection Report 50-400/81-14, 50-401/81-14, 50-402/81-14, and 50-403/81-14, dated August 5 (or 25th), 1981, transmitted a Notice of Violation to CP&L involving "failure to provide records of inspection and monitoring or work performance."

12. NRC Inspection Report 50-400/81-13, 50-401/81-13, 50-402/81-13, and 50-403/81-13, dated August 13, 1981, transmitted a Notice of Violation to CP&L involving "Inadequate Measures to Control Preservation of Safety Related Materials and Equipment."

13. NRC Inspection Report 50-400/81-15, 50-401/81-15, 50-402/81-15, and 50-403/81-15, dated September 14, 1981, transmitted a Notice of Violation to CP&L involving "failure to follow procedure for inspection of fuel pool liner welding."

14. The Alternative Plan, at best, provides assurance that the condition of the Unit 2 spent fuel pool cooling system when the facility was cancelled in December 1983 satisfied the quality standards specified in 10 CFR 50.55a. The NRC inspection reports cited in paragraphs 11 and 13 suggest that these quality standards may not have been met in December 1983. In any case, the Alternative Plan provides no assurance that the spent fuel pool cooling system has not deteriorated since that time. The NRC inspection reports cited in paragraphs 10 and 12 suggest that CP&L had problems protecting against deterioration <u>before</u> Unit 2 was cancelled. In addition, the Alternative Plan contains no provisions to verify that deterioration has not occurred.

11. Nuclear industry experience, as evidence by the cited NRC documents, clearly indicates that installed equipment can deteriorate if not properly maintained. CP&L proposes to use the Unit 2 spent fuel pool cooling system to remove the decay heat from irradiated fuel – a vital safety function that cannot be performed using only the Unit 1 spent fuel pool cooling system. Thus, it is my professional opinion that the failure of the Alternative Plan to provide reasonable assurance against possible deterioration of the installed Unit 2 spent fuel pool cooling system represents a undue challenge to the proposed use of this system.

12. Because it is my professional opinion that the safety concerns addressed in this declaration would be created by the proposed activation the Unit 2 spent fuel pool

cooling system at the Harris Nuclear Plant, I am also of the professional opinion, and do so state here, that the risk to the general public could be increased by the proposed activity, and that the risks and potential are foreseeable, not highly speculative, and potentially significant. Therefore, they should be taken very seriously.

I declare under penalty of perjury that the foregoing facts are true and correct to the best of my knowledge, and the foregoing opinions are based as my best professional judgement.

Executed March 31, 1999.

David A. Lochbaum

Exhibit A: Resume of David A. Lochbaum

Experience Summary

10/96 to date *Nuclear Safety Engineer, Union of Concerned Scientists*

Responsible for directing UCS's nuclear safety program, for monitoring developments in the nuclear industry, for serving as the organization's spokesperson on nuclear safety issues, and for initiating action to correct safety concerns.

11/87 to 09/96 *Senior Consultant, Enercon Services, Inc.*

Responsible for developing the conceptual design package for the alternate decay heat removal system, for closing out partially implemented modifications, reducing the backlog of engineering items, and providing training on design and licensing bases issues at the Perry Nuclear Power Plant.

Responsible for developing a topical report on the station blackout licensing bases for the Connecticut Yankee plant.

Responsible for vertical slice assessment of the spent fuel pit cooling system and for confirmation of licensing commitment implementation at the Salem Generating Station.

Responsible for developing the primary containment isolation devices design basis document, reviewing the emergency diesel generators design basis document, resolving design document open items, and updating design basis documents for the James A. FitzPatrick Nuclear Power Plant.

Responsible for the design review of balance of plant systems and generating engineering calculations to support the Power Uprate Program for the Susquehanna Steam Electric Station.

Responsible for developing the reactor engineer training program, revising reactor engineering technical and surveillance procedures and providing power manuevering recommendations at the Hope Creek Generating Station.

Responsible for supporting the lead BWR/6 Technical Specification Improvement Program and preparing licensing submittals for the Grand Gulf Nuclear Station.

03/87 to 08/87 *System Engineer, General Technical Services*

Responsible for reviewing the design of the condensate, feedwater and raw service systems for safe shutdown and restart capabilities for the Browns Ferry Nuclear Plant.

08/83 to 02/87 *Senior Engineer, Enercon Services, Inc.*

Responsible for performing startup and surveillance testing, developing core monitoring software, developing the reactor engineer training program, and supervising the reactor engineers and Shift Technical Advisors at the Grand Gulf Nuclear Station.

EXHIBIT 4

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISISON
Before the
ATOMIC SAFETY AND LICENSING BOARD

In the matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY	)	**Docket No. 50-400**
	)	
(Harris Nuclear Plant)	)	**March 31, 1999**

DECLARATION OF DAVID A. LOCHBAUM, NUCLEAR SAFETY ENGINEER, UNION OF CONCERNED SCIENTISTS, CONCERNING TECHNICAL ISSUES AND SAFETY MATTERS INVOLVED IN THE HARRIS NUCLEAR PLANT LICENSE AMENDENT FOR SPENT FUEL STORAGE

I, David A. Lochbaum, make the following declaration:

1. My name is David A. Lochbaum. I reside in the state of Maryland.

2. I am employed by the Union of Concerned Scientists as its nuclear safety engineer. I have been so employed since October 1996. I have the following responsibilities: a) direct and coordinate UCS's nuclear safety program; b) monitor developments in nuclear industry to assess and respond to impact; c) serve as technical authority and spokesperson on nuclear issues; and d) initiate legal action to correct safety problems.

3. The Union of Concerned Scientists, with offices located at 1616 P Street NW Suite 310, Washington, DC 20036, is an independent nonprofit organization dedicated to advancing responsible public policies in areas where technology plays a critical role.

4. I have worked in the field of nuclear engineering since June 1979. I am a graduate of the University of Tennessee with a bachelor of science in nuclear engineering.

5. After receiving my nuclear engineering degree, I went to work for the Georgia Power Company as a junior engineer at their Edwin I. Hatch Nuclear Power Plant. I held various positions in the commercial nuclear power industry over the next 17 years prior to joining UCS. This experience is detailed in the resume attached hereto as Exhibit A.

6. I am the author of *Nuclear Waste Disposal Crisis* (Pennwell Books, Tulsa, January 1996) on the technical problems with spent fuel storage at reactor sites.

7. I have examined the license amendment application dated December 23, 1998, submitted by the Carolina Power & Light Company (CP&L) to the Nuclear Regulatory Commission (NRC) concerning spent fuel storage at the Harris Nuclear Plant and the report prepared by Gordon Thompson of the Institute for Resource and Security Studies titled, "Risks and Alternative Options Associated with Spent Fuel Storage at the Shearon Harris Nuclear Power Plant," dated February 1999. I am familiar with these documents and have relied upon them in formulating the opinions contained in this declaration. I have also examined and am familiar with, for the purposes of preparing this declaration, the applicable federal regulations contained in Title 10 of the Code of Federal Regulations; NRC Information Notice No. 85-30: "Microbiologically Induced Corrosion of Containment Service Water System," dated April 19, 1985; NRC Information Notice No. 85-56: "Inadequate Environment Control For Components and Systems in Extended Storage or Layup," dated July 15, 1985; NRC Information Notice No. 94-38: "Results of

a Special NRC Inspection at Dresden Nuclear Power Station Unit 1 Following a Rupture of Service Water Inside Containment," dated May 27, 1994; NRC Inspection Manual Procedure 92050: "Review of Quality Assurance For Extended Construction Delay;" NRC Inspection Report No. 50-400/80-26, 50-401/80-24, 50-402/80-24, and 50-403/80-24, dated January 2, 1981; letter from Darrell G. Eisenhut, Nuclear Regulatory Commission, to Christopher John Adams, The Orange County Democratic Party, dated June 9, 1981; NRC Inspection Report No. 50-400/81-14, 50-401/81-14, 50-402/81-14, and 50-403/81-14, dated August 5 (or 25th), 1981; NRC Inspection Report No. 50-400/81-13, 50-401/81-13, 50-402/81-13, and 50-403/81-13, dated August 13, 1981; and NRC Inspection Report No. 50-400/81-15, 50-401/81-15, 50-402/81-15, and 50-403/81-15, dated September 14, 1981. I have also relied upon these documents in formulating my opinions as expressed in this declaration.

8. Having examined the relevant documents as mentioned above, it is my professional opinion that CP&L's proposed use of an alternative plan per 10 CFR 50.55a to demonstrate that the Unit 2 fuel pool cooling system was "designed, fabricated, erected, constructed and inspected to quality standards commensurate with the importance of the safety function to be performed"[1] raises significant safety concerns for persons living near the facility. It is also my professional opinion that these significant safety concerns have not been adequately considered in the license amendment application filed by CP&L. These concerns are set forth below. I recommend that they be

[1] 10 CFR 50.55a, Codes and standards.

included in the subject matter of issues to be considered by the Atomic Safety and Licensing Board in the above captioned proceeding.

9. It is my professional opinion that the Alternative Plan, as described in Enclosure 8 to the December 23, 1998, license amendment application submitted by CP&L, is deficient for the following reasons:

(a) CP&L notified the NRC in December 1981 that construction on Units 3 and 4 at the Harris Nuclear Plant had been cancelled. In December 1983, CP&L notified the NRC that Unit 2 had been cancelled. Unit 1 was completed and placed into commercial operation in May 1987.[2]

(b) The Alternative Plan describes the process CP&L proposes to certify the installed portions of the Unit 2 spent fuel pool cooling system at the Harris Nuclear Plant in lieu of the original construction records which purportedly were discarded in September 1993.[3]

(c) The Alternative Plan includes review of available documentation, inspection and/or examination of accessible components, internal (via camera) inspections of selected inaccessible components, and hydrostatic testing.

(d) The Alternate Plan and the license amendment application do <u>not</u> describe any program for proper storage and preservation of materials and components as required by Appendix B to 10 CFR Part 50. Nor do they describe any effort to determine if the installed piping and equipment experienced any deterioration over the many years of non-use since the piping and equipment were installed.

[2] CP&L, Enclosure 8 to December 23, 1998 submittal, pp. 1-2.
[3] CP&L, Enclosure 8 to December 23, 1998 submittal, page 11.

(e) NRC Information Notice No. 85-30 documents a problem experienced at the H B Robinson nuclear plant, also owned and operated by CP&L, during 1984. According to this NRC document, stainless steel piping at the Robinson plant experienced significant corrosion pitting during an outage lasting about one year.

(f) The NRC issued Information Notice No. 85-56 "to alert addressees to problems which can occur if equipment is improperly stored or laid up during construction or extended plant outages. Addressees also are reminded that programs for proper storage and preservation of materials and components are required by NRC regulations (10 CFR 50, Appendix B), even though not specifically addressed as license conditions." According to this NRC document, a heat exchanger at Nine Mile Point Unit 2 was found to have experienced significant corrosion in the eight (8) years it had been stored in-place.

(g) NRC Information Notice No. 94-38 documents a problem at Dresden Unit 1. A water-filled pipe froze and ruptured, causing 55,000 gallons of water to flood the containment building. The NRC reported that the plant's owner had not taken adequate measures to protect the permanently closed facility. The NRC also reported that had a water-filled pipe connected to the spent fuel pool – which was as unprotected as the pipe which failed – ruptured, the water level in the spent fuel pool could have dropped below the top of its irradiated fuel assemblies.

(h) NRC Inspection Procedure 92050 contains guidance for NRC inspectors when auditing nuclear plants encountering extended construction delays. Among other areas, the guidance covers the plant owner's program for the protection and preservation of equipment. The elements of this program include:

1. Protective coverings and coatings,
2. Internal preservation,
3. Dunnage and other supports, and
4. Cleanliness preservation.

The Alternative Plan does not address any of these elements or describe measures taken to protect piping and components during the extended delay between construction and proposed use of this equipment.

10. NRC Inspection Report 50-400/80-26, 50-401/80-24, 50-402/80-24, and 50-403/80-24, dated January 2, 1981, transmitted a Notice of Violation to CP&L involving "failure to store equipment in accordance with instructions to prevent damage or deterioration."

11. NRC Inspection Report 50-400/81-14, 50-401/81-14, 50-402/81-14, and 50-403/81-14, dated August 5 (or 25th), 1981, transmitted a Notice of Violation to CP&L involving "failure to provide records of inspection and monitoring or work performance."

12. NRC Inspection Report 50-400/81-13, 50-401/81-13, 50-402/81-13, and 50-403/81-13, dated August 13, 1981, transmitted a Notice of Violation to CP&L involving "Inadequate Measures to Control Preservation of Safety Related Materials and Equipment."

13. NRC Inspection Report 50-400/81-15, 50-401/81-15, 50-402/81-15, and 50-403/81-15, dated September 14, 1981, transmitted a Notice of Violation to CP&L involving "failure to follow procedure for inspection of fuel pool liner welding."

14. The Alternative Plan, at best, provides assurance that the condition of the Unit 2 spent fuel pool cooling system when the facility was cancelled in December 1983 satisfied the quality standards specified in 10 CFR 50.55a. The NRC inspection reports cited in paragraphs 11 and 13 suggest that these quality standards may not have been met in December 1983. In any case, the Alternative Plan provides no assurance that the spent fuel pool cooling system has not deteriorated since that time. The NRC inspection reports cited in paragraphs 10 and 12 suggest that CP&L had problems protecting against deterioration <u>before</u> Unit 2 was cancelled. In addition, the Alternative Plan contains no provisions to verify that deterioration has not occurred.

11. Nuclear industry experience, as evidence by the cited NRC documents, clearly indicates that installed equipment can deteriorate if not properly maintained. CP&L proposes to use the Unit 2 spent fuel pool cooling system to remove the decay heat from irradiated fuel – a vital safety function that cannot be performed using only the Unit 1 spent fuel pool cooling system. Thus, it is my professional opinion that the failure of the Alternative Plan to provide reasonable assurance against possible deterioration of the installed Unit 2 spent fuel pool cooling system represents a undue challenge to the proposed use of this system.

12. Because it is my professional opinion that the safety concerns addressed in this declaration would be created by the proposed activation the Unit 2 spent fuel pool

cooling system at the Harris Nuclear Plant, I am also of the professional opinion, and do so state here, that the risk to the general public could be increased by the proposed activity, and that the risks and potential are foreseeable, not highly speculative, and potentially significant. Therefore, they should be taken very seriously.

I declare under penalty of perjury that the foregoing facts are true and correct to the best of my knowledge, and the foregoing opinions are based as my best professional judgement.

Executed March 31, 1999.

David A. Lochbaum

David A. Lochbaum

Experience Summary (continued)

10/81 to 08/83 *Reactor Engineer / Shift Technical Advisor, Tennessee Valley Authority*

Responsible for performing core management functions, administering the nuclear engineer training program, maintaining ASME Section XI program for the core spray and CRD systems, and covering STA shifts at the Browns Ferry Nuclear Plant.

06/81 to 10/81 *BWR Instructor, General Electric Company*

Responsible for developing administrative procedures for the Independent Safety Engineering Group (ISEG) at the Grand Gulf Nuclear Station.

01/80 to 06/81 *Reactor Engineer / Shift Technical Advisor, Tennessee Valley Authority*

Responsible for directing refueling floor activities, performing core management functions, maintaining ASME Section XI program for the RHR system, providing power manuevering recommendations and covering STA shifts at the Browns Ferry Nuclear Plant.

06/79 to 12/79 *Junior Engineer, Georgia Power Company*

Responsible for completing pre-operational testing of the radwaste solidification systems and developing design change packages for modifications to the liquid radwaste systems at the Edwin I. Hatch Nuclear Plant.

Education

June 1979 Bachelor of Science in Nuclear Engineering, The University of Tennessee at Knoxville

May 1980 Certification, Interim Shift Technical Advisor, TVA Browns Ferry Nuclear Plant

April 1982 Certification, Shift Technical Advisor, TVA Browns Ferry Nuclear Plant

Professional Affiliations

Member, American Nuclear Society (since 1978).

SR/CNEAF/96-01

Spent Nuclear Fuel Discharges from U.S. Reactors 1994

EXHIBIT 5

February 1996

Energy Information Administration
Office of Coal, Nuclear, Electric and Alternate Fuels
Analysis and Systems Division
U.S. Department of Energy
Washington, DC 20585

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Released for Printing: February 9, 1996

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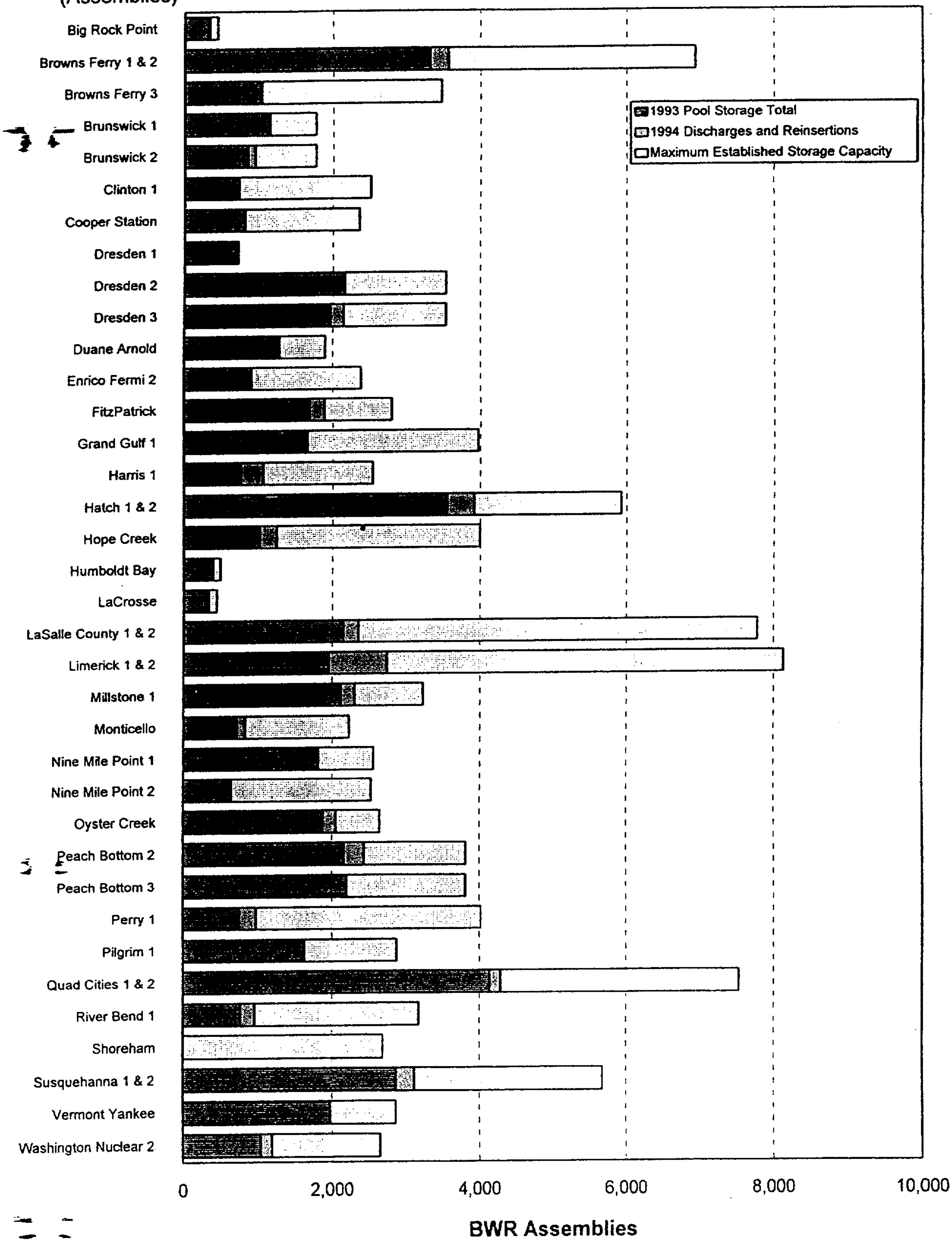
Table 2. Reactor Spent Fuel Discharge and Storage Activity for 1994

Electric Utility Name	Reactor Name	Reactor Type	Pool Site ID	1993 Assembly Total	1994 Assembly Discharges	1994 Assembly Reinsertions	Other	1994 Assembly Total
Alabama Power Company	Farley 1	PWR	0101	655	61			716
	Farley 2	PWR	0102	558				558
Arizona Public Service Company	Palo Verde 1	PWR	0301	368				368
	Palo Verde 2	PWR	0302	384				384
	Palo Verde 3	PWR	0303	284	124	28		380
Arkansas Power and Light Company	Arkansas Nuclear 1	PWR	0401	684				684
	Arkansas Nuclear 2	PWR	0402	564	73	1		636
Baltimore Gas and Electric Company	Calvert Cliffs 1	PWR	0501	1,450	89	1	[a]-122	1,394
	Calvert Cliffs 2	PWR		0			[a]-22	
	Dry Storage	PWR	0501D	48			[a]144	192
Boston Edison Company	Pilgrim 1	BWR	0601	1,628				1,628
Carolina Power and Light Company	Brunswick 1	BWR	0701	1,146			[b]-204	942
	Brunswick 1	PWR	0701	160				160
	Brunswick 2	BWR	0702	841	152		[b]-102	891
	Brunswick 2	PWR	0702	144				144
	Harris 1	PWR	0703	448	61	9		500
	Harris 1	BWR	0703	753			[b]306	1,059
	Robinson 2	PWR	0705	240				240
	Dry Storage	PWR	0705D	56				56
Cleveland Electric Illuminating Company	Perry 1	BWR	0901	748	228	4		972
		BWR	0902					
Commonwealth Edison Company	Braidwood 1	PWR	1001	488	92			668
	Braidwood 2	PWR			92	4		
	Byron 1	PWR	1003	772	92			864
	Byron 2	PWR						
	Dresden 1	BWR	1005	683				683
	Dresden 2	BWR	1006	2,162				2,162
	Dresden 3	BWR	1007	1,968	180			2,148
	LaSalle County 1	BWR	1008	2,152	208			2,360
	LaSalle County 2	BWR		0				
	Quad Cities 1	BWR	1010	4,140	144			4,284
	Quad Cities 2	BWR		0				
	Zion 1	PWR	1012	1,684				1,684
	Zion 2	PWR						

See footnotes at end of table.

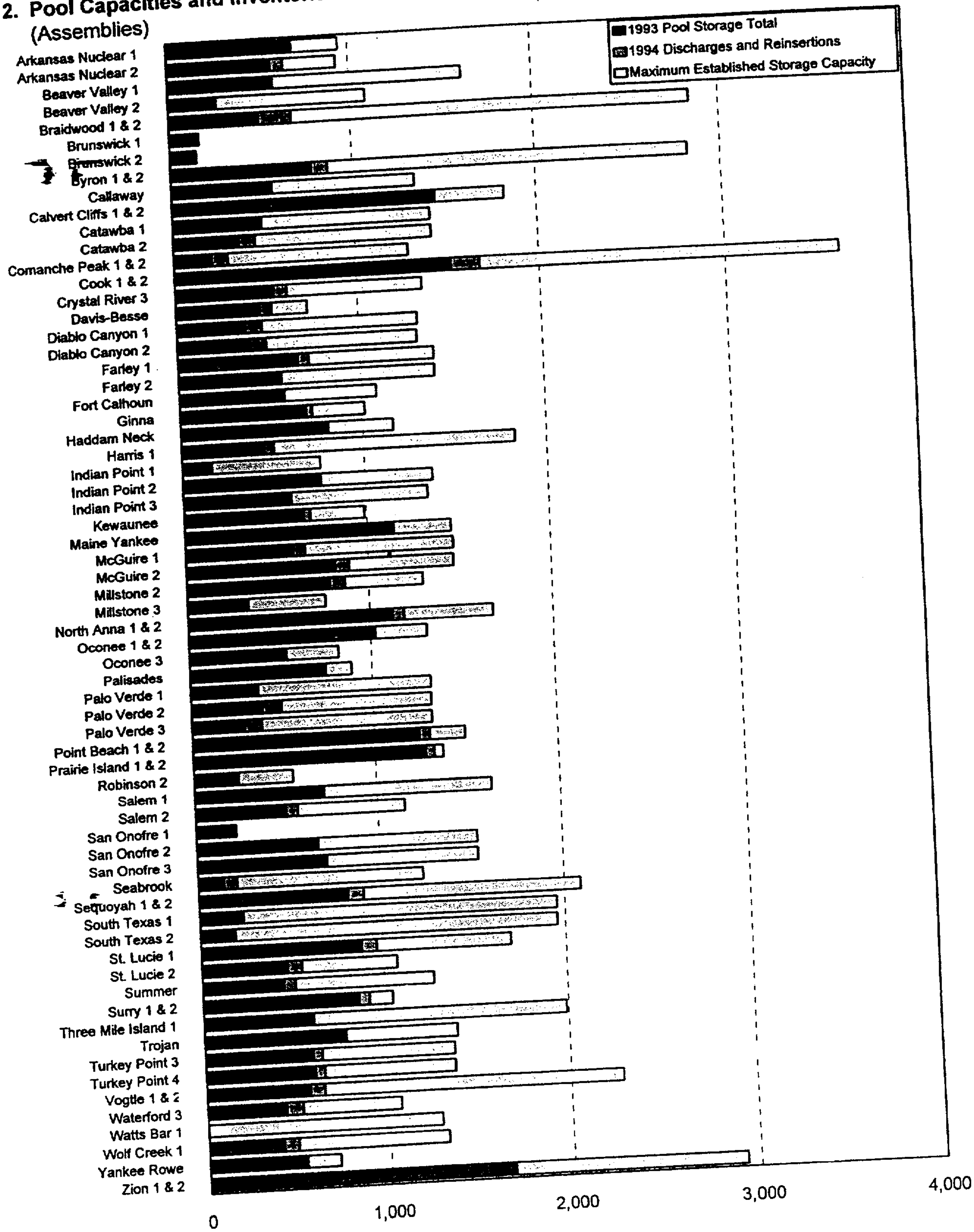
Figure 1. Pool Capacities and Inventories for Boiling-Water Reactors (Assemblies)



Notes: Data includes all boiling-water reactors (BWR's) reported on the Form RW-859. Number of 1994 Discharges and Reinsertions does not include intrautility transfers. See Table 2. Values of 1994 Discharges and Reinsertions for Limerick 1 & 2 and Shoreham reflect the transfer of 254 assembliesfrom Shoreham to Limerick 1 in 1994. See Technical Note 14 in Appendix E.

Source: Energy Information Administration, Form RW-859, "Nuclear Fuel Data" (1994).

Figure 2. Pool Capacities and Inventories for Pressurized-Water Reactors (Assemblies)



Notes: Data includes all pressurized-water reactors (PWR's) reported on the Form RW-859. Number of 1994 Discharges and Reinsertions does not include intrautility transfers. See Table 2.
Source: Energy Information Administration, Form RW-859, "Nuclear Fuel Data" (1994).

Table 3. Temporarily Discharged Assemblies

Electric Utility Name	Reactor Name	Reactor Type	Pool Site ID	Temporarily Discharged Assemblies Through 1994	Through 1993	Increase/ Reduction
Arizona Public Service Company	Palo Verde 3	PWR	0303	1	1	0
Carolina Power and Light Company	Brunswick 1	BWR	0701	1	1	0
Cleveland Electric Illuminating Company	Perry 1	BWR	0901	0	4	-4
Consolidated Edison Company of New York	Indian Point 2	PWR	1102	8	8	0
Consumers Power Company	Big Rock Point	BWR	1201	1	0	1
	Palisades	PWR	1204	1	1	0
Detroit Edison Company	Enrico Fermi 2	BWR	1402	40	40	0
Duquesne Light Company	Beaver Valley 1	PWR	1601	13	13	0
Florida Power Corporation	Crystal River 3	PWR	1701	5	5	0
GPU Nuclear Corporation	Three Mile Island 1	PWR	1901	23	23	0
Houston Lighting and Power Company	South Texas 1	PWR	2201	7	7	0
Maine Yankee Atomic Power Company	Maine Yankee	PWR	2801	18	18	0
New York Power Authority	Indian Point 3	PWR	3902	2	2	0
PECO Energy Company	Limerick 1	BWR	3701	[a]504	306	198
Public Service Electric and Gas Company	Salem 1	PWR	4202	23	23	0
	Salem 2	PWR	4203	34	35	-1
Tennessee Valley Authority	Browns Ferry 2	BWR	4803	80	80	0
Toledo Edison Company	Davis-Besse	PWR	5001	2	6	-4
Union Electric Company	Callaway	PWR	5101	2	2	0
Virginia Power	North Anna 1	PWR	5201	19	25	-6
	Surry 1	PWR	5203	10	6	4
Wisconsin Public Service Corporation	Kewaunee	PWR	5501	4	4	0
Total				798	610	188

[a]A total of 560 temporarily discharged assemblies were shipped from Long Island Power Authority's Shoreham plant to Limerick 1. Of these, a total of 56 temporarily discharged assemblies were reinserted in core at Limerick 1. See Technical Note 14 in Appendix E.

PWR = Pressurized-water reactor; BWR = Boiling-water reactor.

Note: Changes in number of temporarily discharged assemblies are due to discharge of additional temporarily discharged assemblies, reinsertion of previously discharged assemblies, and/or change in status of previously discharged assemblies.

Source: Energy Information Administration, Form RW-859, "Nuclear Fuel Data" (1994).

Table 4. Nuclear Power Plant Data as of December 31, 1994

Electric Utility Name	Reactor Name	State	Reactor Type	Vendor[a]	Capacity (net MWe)[b]	Core Size (number of assemblies)	Date of Operation (year)[c]	License Expiration (year)	Loss of Ability to Operate (year)[d]	Actual or Projected Retirement (year)
Alabama Power Company . .	Farley 1	AL	PWR	WE	815	157	1977	2017	2010	2017
	Farley 2	AL	PWR	WE	825	157	1981	2021	2013	2021
Arizona Public Service Company	Palo Verde 1	AZ	PWR	CE	1,270	241	1985	2024	2005	2024
	Palo Verde 2	AZ	PWR	CE	1,270	241	1986	2025	2005	2025
	Palo Verde 3	AZ	PWR	CE	1,270	241	1987	2027	2006	2027
Arkansas Power and Light Company	Arkansas Nuclear 1	AR	PWR	B&W	836	177	1974	2014	1996	2014
	Arkansas Nuclear 2	AR	PWR	CE	858	177	1978	2018	1997	2018
Baltimore Gas and Electric Company	Calvert Cliffs 1	MD	PWR	CE	835	217	1974	2014	2007	2014
	Calvert Cliffs 2	MD	PWR	CE	840	217	1976	2016	2007	2016
Boston Edison Company . . .	Pilgrim 1	MA	BWR	GE	665	580	1972	2012	2003	2012
Carolina Power and Light Company	Brunswick 1	NC	BWR	GE	767	560	1976	2016	2002	2016
	Brunswick 2	NC	BWR	GE	754	560	1974	2014	2003	2014
	Harris 1	NC	PWR	WE	860	157	1987	2026	2026	2026
	Robinson 2	SC	PWR	WE	683	157	1970	2010	2004	2010
Cleveland Electric Illuminating Company	Perry 1	OH	BWR	GE	1,169	748	1986	2026	2013	2026
Commonwealth Edison Company	Braidwood 1	IL	PWR	WE	1,090	193	1987	2026	2012	2028
	Braidwood 2	IL	PWR	WE	1,090	193	1988	2027	2012	2028
	Byron 1	IL	PWR	WE	1,120	193	1985	2024	2011	2025
	Byron 2	IL	PWR	WE	1,120	193	1987	2026	2011	2027
	Dresden 1	IL	BWR	GE	200	464	1959	1996	SD	1984
	Dresden 2	IL	BWR	GE	772	724	1969	2006	2001	2010
	Dresden 3	IL	BWR	GE	773	724	1971	2011	2002	2013
	LaSalle County 1	IL	BWR	GE	1,048	764	1982	2022	2013	2024
	LaSalle County 2	IL	BWR	GE	1,048	764	1984	2023	2013	2024
	Quad Cities 1	IL	BWR	GE	769	724	1972	2012	2009	2013
	Quad Cities 2	IL	BWR	GE	769	724	1972	2012	2009	2013
	Zion 1	IL	PWR	WE	1,040	193	1973	2013	2006	2013
	Zion 2	IL	PWR	WE	1,040	193	1973	2013	2006	2014

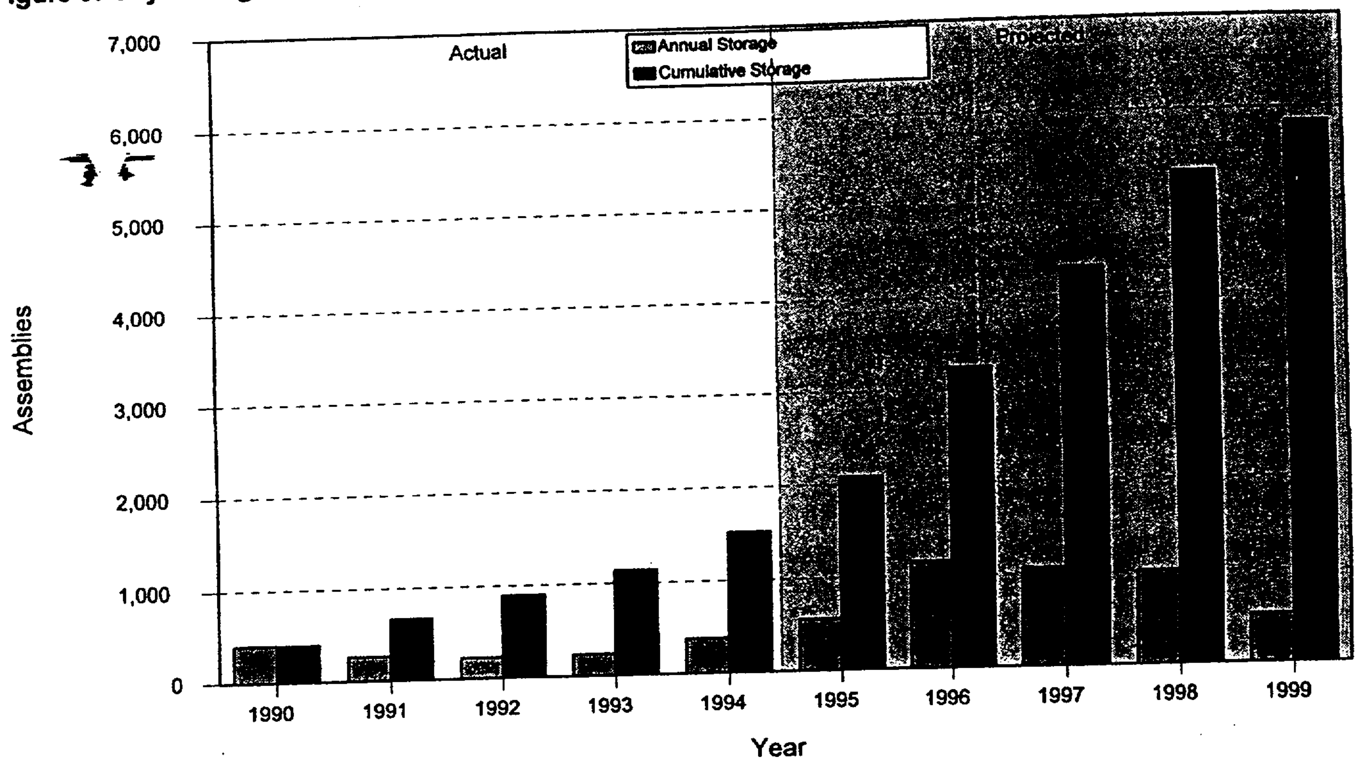
See footnotes at end of table.

Table 10. Site Capacities and Inventories at Nuclear Power Plants as of December 31, 1994

Electric Utility Name	Reactor/Storage Site Name	Reactor Type	Pool Site ID	Licensed Storage Capacity[a]	Maximum Established Storage Capacity[b]	Current Inventory[c]	Initial Uranium Content of Pool Inventory (MTU)	Number of Temporarily Discharged Assemblies[d]
Alabama Power Company . . .	Farley 1	PWR	0101	1,407	1,407	716	330.1	0
	Farley 2	PWR	0102	1,407	1,407	558	257.4	0
Arizona Public Service Company	Palo Verde 1	PWR	0301	665	1,323	368	151.7	0
	Palo Verde 2	PWR	0302	665	1,323	384	156.5	0
	Palo Verde 3	PWR	0303	665	1,322	380	156.4	1
Arkansas Power and Light Company	Arkansas Nuclear 1	PWR	0401	968	948	684	316.9	0
	Arkansas Nuclear 2	PWR	0402	988	933	636	263.9	0
Baltimore Gas and Electric Company	Calvert Cliffs 1	PWR	0501	1,830	1,830	1,394	533.9	0
	Calvert Cliffs 2	PWR		0	0	0	0.0	0
	Dry Storage	PWR	0501D	2,880	1,152	192	73.9	0
Boston Edison Company	Pilgrim 1	BWR	0601	2,320	2,875	1,628	301.6	0
Carolina Power and Light Company	[e]Brunswick 1	BWR	0701	1,803	1,767	942	174.4	1
	[e]Brunswick 1	[f]PWR	0701	160	160	160	71.3	0
	[e]Brunswick 2	BWR	0702	1,839	1,767	891	164.8	0
	[e]Brunswick 2	[f]PWR	0702	144	144	144	65.5	0
	[g]Harris 1	PWR	0703	4,184	1,832	500	221.2	0
	[g]Harris 1	BWR	0703	5,808	2,541	1,059	195.4	0
	Robinson 2	PWR	0705	544	537	240	102.0	0
	Dry Storage	PWR	0705D	56	56	56	24.1	0
Cleveland Electric Illuminating Company	Perry 1	BWR	0901	4,020	4,020	972	177.9	0
		BWR	0902	0	0	0	0.0	0

See footnotes at end of table.

Figure 9. Dry Storage Inventories and Projections



Source: Energy Information Administration, Form RW-859, "Nuclear Fuel Data" (1994).

Arkansas Power and Light Company

In 1994, the Arkansas Power and Light Company's Arkansas Nuclear 1 & 2 plants finalized a contractual agreement with Sierra Nuclear for 14 VSC-24 casks that can hold up to 24 PWR assemblies.

Construction of the concrete casks began in October 1994. The cask storage will be on a concrete pad located within the existing security protected area at the Arkansas Nuclear plants. The pad is designed to hold 26 casks, but can be expanded to provide space for an additional 50 casks. Existing rail lines and a new rail car specifically designed for the VSC will transport the casks from the plant's Auxiliary Building to the storage pad.

Baltimore Gas and Electric Company

The ISFSI at Baltimore Gas and Electric Company's Calvert Cliffs station is the NUHOMS-24P. The Calvert Cliffs ISFSI has been designed as a life-of-plant storage facility. The ISFSI will have the capacity to store all spent fuel discharged from Calvert Cliffs 1 & 2, beyond the spent fuel pool capacity, up to the 40-year plant life, if necessary. The exact capacity needed is uncertain, and to limit capital investment until necessary, the ISFSI will be constructed in up to five phases. The ISFSI required the preparation of a 10 CFR 72 License Application, Safety Analysis Report, Environmental Report, and a Security Plan for NRC review and approval. The license material was prepared and submitted to the NRC in December 1989. Construction of the ISFSI west of the plant began in April 1991 after NRC approved the Environmental Report. The facility and its pre-operational testing were completed in October 1992. The ISFSI was licensed by the NRC on November 25, 1992.

The license allows Baltimore Gas and Electric Company to place as many as 2,880 assemblies in casks to be placed in ISFSI's. Each NUHOMS cask at Calvert Cliffs can hold 24 assemblies, and there are currently 120 planned storage modules. On November 30, 1993, the dry storage facility became fully operational with the successful loading of the first cask of fuel. As of September 1995, a total of 240 assemblies were stored in 10 modules.

Carolina Power and Light Company

The ISFSI for Carolina Power and Light Company's Robinson 2 plant is composed of 8 NUHOMS-7P horizontal storage modules (HSM's). Each HSM is a steel-reinforced concrete structure which holds 7 intact assemblies in each module. The ISFSI was licensed by the NRC in August 1986 to hold 56 assemblies, and became operational in March 1989. The Carolina Power and Light Company also applied to the NRC for a license for an ISFSI to be built at its Brunswick plant. The ISFSI at the Brunswick plant will be used only as a backup if shipping of spent nuclear fuel to the Harris plant is prohibited.

Consumers Power Company

In April 1993, the NRC granted a Certificate of Compliance to the VSC-24 cask by Sierra Nuclear Corporation and approved use of dry-cask spent fuel storage at Consumers Power Company's Palisades plant. The approval was challenged by the Michigan Attorney General and a citizen organization, the Lake Michigan Federation, on the grounds that the process should have entailed a full environmental impact statement, rather than the less elaborate environmental assessment.

Palisades plant personnel started loading casks on May 7, 1993, and by May 19, 1993, two casks, containing 24 spent fuel assemblies, were welded shut and placed on the storage pad. By May 1995, 11 additional casks were loaded, for a total of 13 loaded casks. The design allows for as many as 25 casks to be used at the plant -- enough to last Palisades through the end of its current licensed life, in 2007.

On August 1, 1994, Consumers Power Company notified the NRC of its plans to unload and replace 1 of the dry storage casks in use at the Palisades plant. Although no leaks were detected, the utility found indications of minor flaws in the welds of the VSC-24 cask during its review of the manufacturer's quality assurance program. The utility stated that even though there was no actual health, engineering, or operational requirements, the cask was to be replaced.

In January 1995, the U.S. Circuit Court of Appeals declined to order the NRC to conduct a full hearing before allowing Consumers Power to continue using dry storage at the Palisades plant.[1] The three judge panel said that the NRC had taken all the necessary steps to safeguard the environment, even though it had not prepared the site-specific analysis. The Appeals Court upheld the NRC's contention that the pad site was included in the environmental impact statement and that the pad site was acceptable for use.

In July 1995, the U.S. Supreme Court denied an appeal by the Michigan Attorney General and two environmental organizations challenging the licensing procedures used in approving the ISFSI2.[2] The denial upheld the January 1995 decision by the U.S. Circuit Court of Appeals, thus allowing for future use of the ISFSI for spent fuel storage.

Duke Power Company

Duke Power Company received its Oconee site license from the NRC in January 1990 for 88 NUHOMS-24P module. Each module is designed to store 24 pressurized-water reactor (PWR) assemblies; therefore, the maximum capacity is 2,112 assemblies. The first 20 modules were completed in 1990 and the second set of 20 in 1992. The modules were loaded with fuel as follows: 4 modules in 1990, 9 modules in 1991, 7 modules in 1992, 4 modules in 1993, and 5 modules in 1994. As of September 1995, a total of 816 assemblies in 34 modules are stored in Oconee's ISFSI. Duke Power plans to load an average of 5 modules each year from 1996 to 1998.

GPU Nuclear Corporation

GPU Nuclear Corporation contracted VECTRA Technologies, Inc., to engineer, license, and construct a spent fuel storage system for the Oyster Creek nuclear plant. The agreement includes the design and construction of concrete modules for use in storing the plant's spent fuel on-site. The facility will employ the NUHOMS-52B module design. VECTRA Technologies completed engineering and licensing work in 1994 and began delivering the fuel-storage equipment in 1995. Initial loading of the facility is expected in February 1996. Plans are for a total capacity of 20 storage modules (1,040 assemblies). This will be the first dry fuel storage project for BWR fuel in the United States.

New York Power Authority

New York Power Authority's FitzPatrick plant contracted with VECTRA Technologies, Inc. for the design and implementation of a 34 module NUHOMS-52B ISFSI facility. As the NUHOMS-52B is a NRC-approved dry storage system, FitzPatrick's ISFSI does not require a site-specific license. Phase I, which included the design criteria, conceptual design, site selection, and geotechnical investigation, was completed in October 1994.

Northern States Power Company

The decision by the Minnesota Public Utilities Commission to allow 17 containers for aboveground spent fuel storage at Northern States Power Company's Prairie Island site was granted June 26, 1992.[3] The number of containers granted for use was considerably less than Northern States Power

[1] 42 F.3d 1501, *; U.S. App. LEXIS 371, **1; 1995 FED App. 0013P (6th Cir.)
[2] "The U.S. Supreme Court Denied An Appeal," *Nuclear News* (August 1995), p.84.
[3] 131 Pub. Util. Rep 4th (PUC) 315 (Minn. PUC 1992)

Table 13. Independent Spent Fuel Storage Installation (ISFSI) Data

Electric Utility Name	Reactor	Date License Issued/ Submitted	Vendor	Storage Type	Model	Planned Capacity[a] No. of Modules	Assemblies per Module	Total
Arkansas Power and Light Company	Arkansas Nuclear 1 & 2	[b]General	Sierra Nuclear Corporation	Concrete Cask	VSC-24	14	24	336
Baltimore Gas and Electric Company	Calvert Cliffs 1 & 2	11/92	VECTRA Technologies, Inc.	Concrete Module	NUHOMS-24P	120	24	2,880
Carolina Power and Light Company	Brunswick 1 & 2	[c]05/89	VECTRA Technologies, Inc.	Concrete Module	NUHOMS-7P	44	7	308
Carolina Power and Light Company	Robinson 2	08/86	VECTRA Technologies, Inc.	Concrete Module	NUHOMS-7P	8	7	56
Consumers Power Company	Palisades	[c]03/90	Sierra Nuclear Corporation	Concrete Cask	VSC-24	24	24	576
Duke Power Company	Oconee 1,2,3	01/90	VECTRA Technologies, Inc.	Concrete Module	NUHOMS-24P	88	24	2,112
GPU Nuclear Corporation	Oyster Creek	[b]General	VECTRA Technologies, Inc.	Concrete Module	NUHOMS-52B	20	52	1,040
New York Power Authority	FitzPatrick	[b]General	VECTRA Technologies	Concrete Module	NUHOMS-52B	34	52	1,768
Northern States Power Company	Prairie Island 1 & 2	10/93	Transnuclear, Inc.	Metal Cask	TN-40	17	40	680
Pennsylvania Power and Light Company	Susquehanna 1 & 2	[b]General	VECTRA Technologies, Inc.	Concrete Module	NUHOMS-52B	105	52	5,460
Portland General Electric Company	Trojan	[c]02/96	Sierra Nuclear Corporation	Concrete Module	TRANSTOR	36	24	864
Public Service Company of Colorado	Fort St. Vrain	11/91	Foster Wheeler Energy Applications, Inc.	Concrete Vault	Modular Dry Storage	6	244	1,464

See footnotes at end of table.

Table 16. Canisters Containing Spent Fuel as of December 31, 1994

Electric Utility Name	Reactor Name	Reactor Type	Pool Site ID	Fuel Canisters Containing: Intact Assemblies	Rods and Pieces[a]	Consolidated Assemblies	Number of Consolidated Assemblies	Unknown Contents[b]	Total Fuel Canisters[c]
Alabama Power Company	Farley 1	PWR	0101	0	0	0	0	0	0
	Farley 2	PWR	0102	0	0	0	0	0	0
Arizona Public Service Company	Palo Verde 1	PWR	0301	0	3	0	0	0	3
	Palo Verde 2	PWR	0302	0	2	0	0	0	2
	Palo Verde 3	PWR	0303	0	1	0	0	0	1
Arkansas Power and Light Company	Arkansas Nuclear 1	PWR	0401	0	0	0	0	0	0
	Arkansas Nuclear 2	PWR	0402	0	1	0	0	0	1
Baltimore Gas and Electric Company	Calvert Cliffs 1 & 2	PWR	0501	0	0	0	0	0	0
Boston Edison Company	Pilgrim 1	BWR	0601	1	0	0	0	0	1
Carolina Power and Light Company	Brunswick 1	BWR	0701	0	1	0	0	0	1
	Brunswick 2	BWR	0702	0	1	0	0	0	1
	Harris 1	PWR	0703	0	1	0	0	0	1
	Robinson 2	PWR	0705	0	1	0	0	0	1
Cleveland Electric Illuminating Company	Perry 1	BWR	0901	2	1	0	0	0	3
Commonwealth Edison Company	Braidwood 1 & 2	PWR	1001	0	1	0	0	0	1
	Byron 1 & 2	PWR	1003	4	0	0	0	0	4
	Dresden 1	BWR	1005	0	1	0	0	0	1
	Dresden 2	BWR	1006	0	1	0	0	0	1
	Dresden 3	BWR	1007	0	2	0	0	0	2
	LaSalle County 1 & 2	BWR	1008	0	1	0	0	0	1
	Quad Cities 1 & 2	BWR	1010	0	3	0	0	0	3
	Zion 1 & 2	PWR	1012	1	2	0	0	0	3
Consolidated Edison Company of New York	Indian Point 1	PWR	1101	0	0	0	0	0	0
	Indian Point 2	PWR	1102	0	0	0	0	0	0
Consumers Power Company	Big Rock Point	BWR	1201	0	7	0	0	0	7
	Palisades	PWR	1204	0	7	0	0	0	7

See footnotes at end of table.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD '99 APR -7 P3:27

In the Matter of CAROLINA POWER & LIGHT (Shearon Harris Nuclear Power Plant)	Docket No. 50-400 -OLA ASLBP No. 99-762-02-LA

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RULEMAKINGS AND
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CERTIFICATE OF SERVICE

I certify that on April 5, 1999, copies of the foregoing Orange County's Supplemental Petition to Intervene were served on the following by e-mail and/or first class mail as indicated below:

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