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, 0.65×10^5 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 0.65×10^5 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 2.3×10^7 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 7.1×10^7 Curies (2,600,000 TBq) of cesium-137, with a mass of 790 kilograms.

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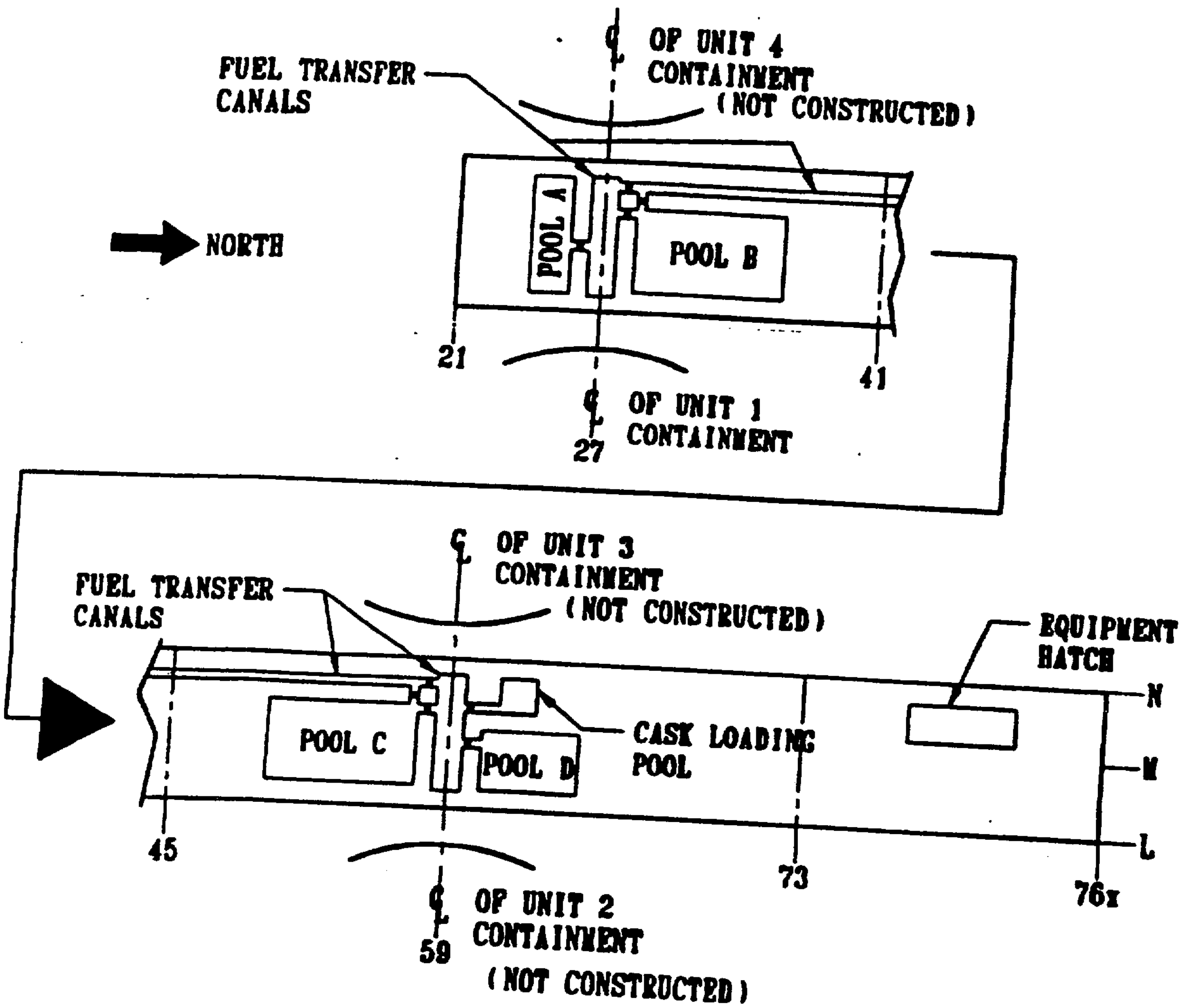
14 The ratio of 1/4 derives from the parameters shown in the license amendment application, Enclosure 7, page 5-15.
15 J Scarola of CP&L, presentation to Orange County Board of Commissioners, 9 February 1999.

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.

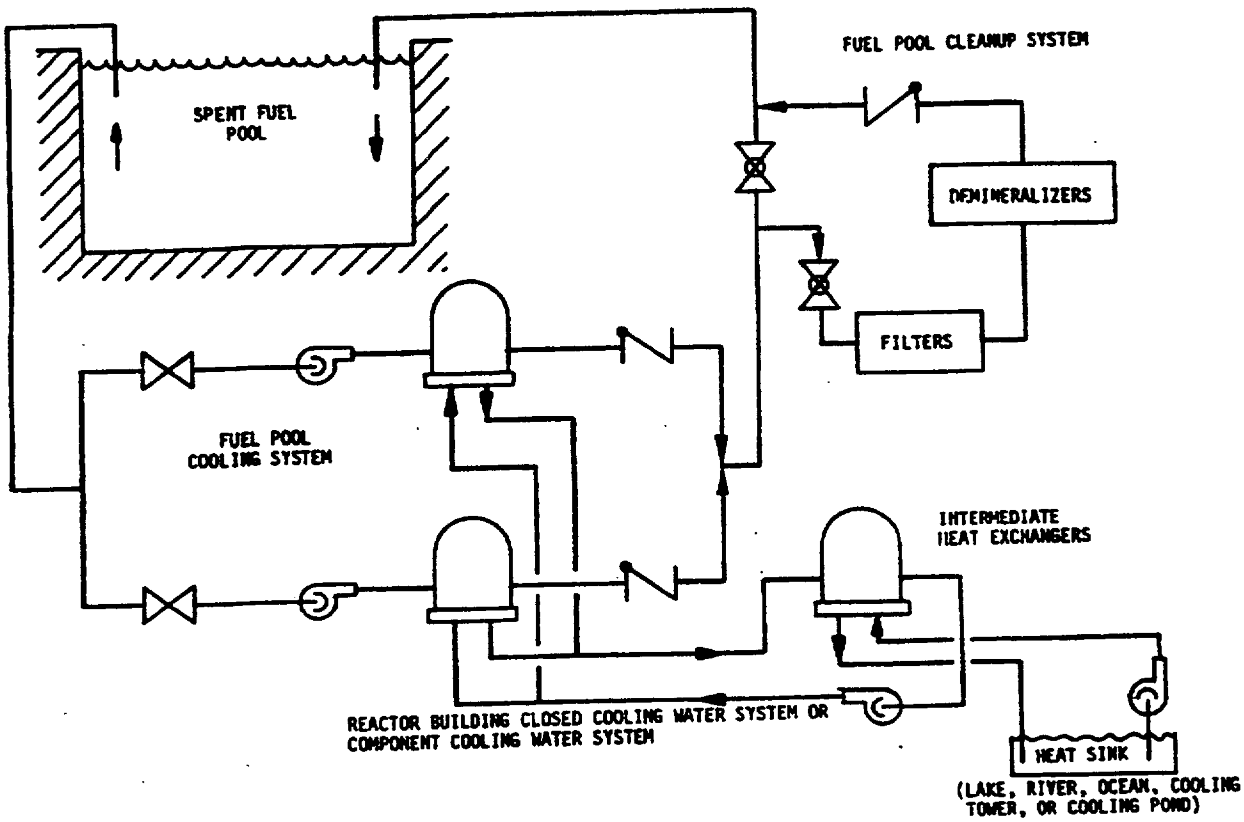
16 US Nuclear Regulatory Commission, Final Environmental Statement Related to the Operation of Shearon Harris Nuclear Power Plant Units 1 and 2, NUREG-0972, October 1983.



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

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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 1×10^{-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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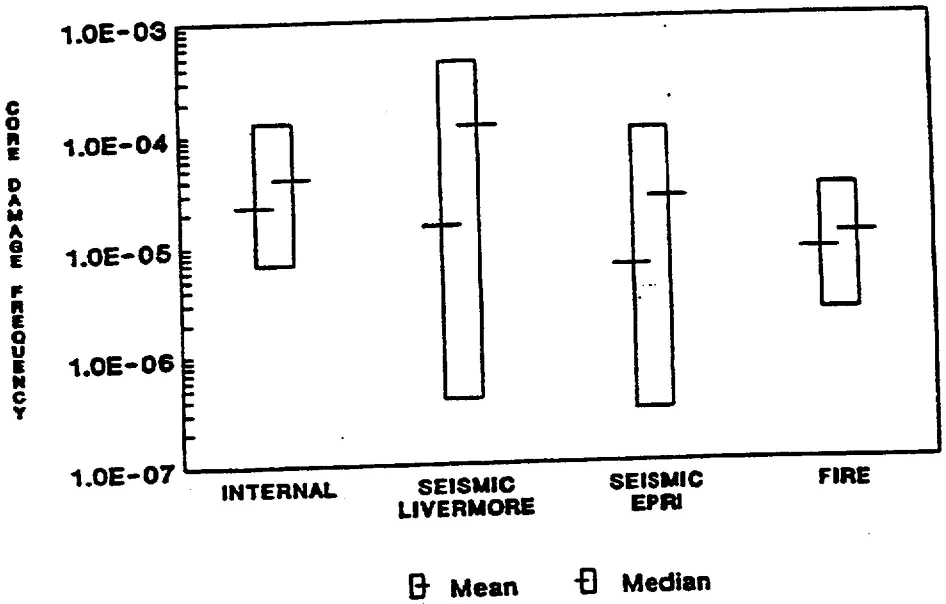
[11] Release categories involving significant containment failure or bypass are, in descending order of estimated probability, RC-4, RC-5, RC-6, RC-1B, RC-4C and RC-3. Each of these categories involves a 100 percent release of noble gases. The CsI release fraction ranges from .001 percent (RC-6) to 59 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.

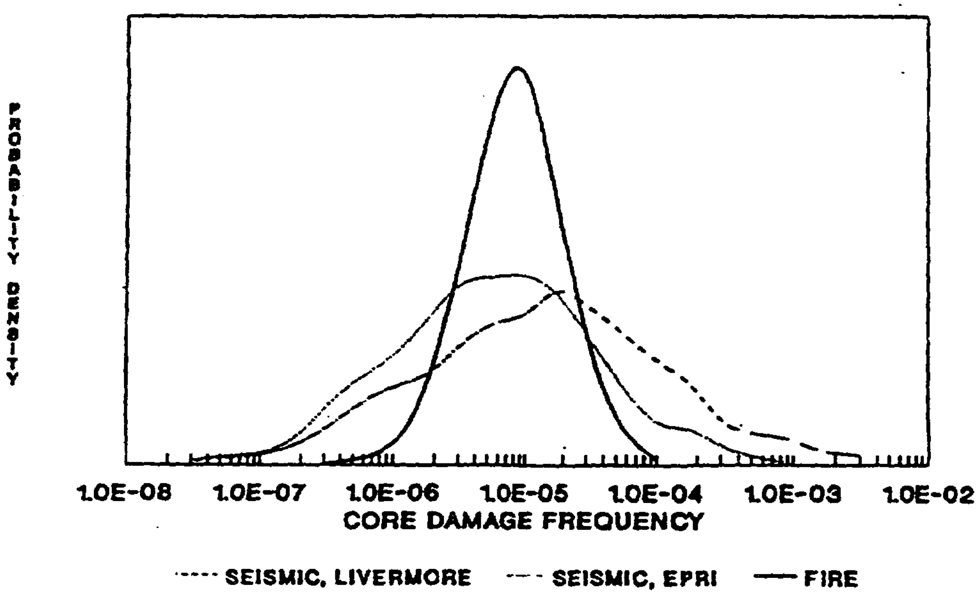


—Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Source: NUREG-1150

Figure B-1

Estimated core damage frequency for the Surry PWRs

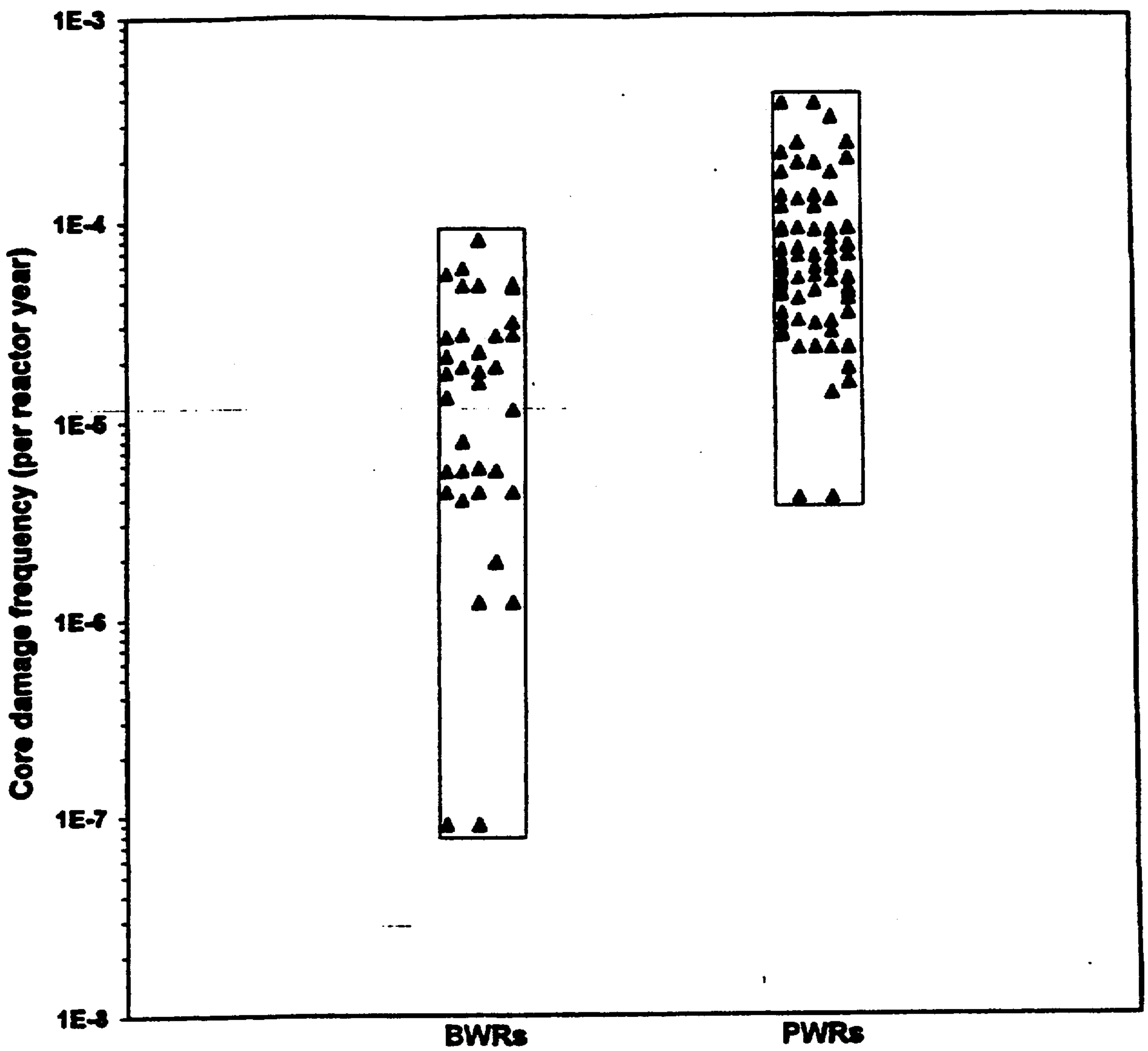


Note: As discussed in Reference 8.7, core damage frequencies below 1E-5 per reactor year should be viewed with caution because of the remaining uncertainties in PRA (e.g., events not considered).

Source: NUREG-1150

Figure B-2

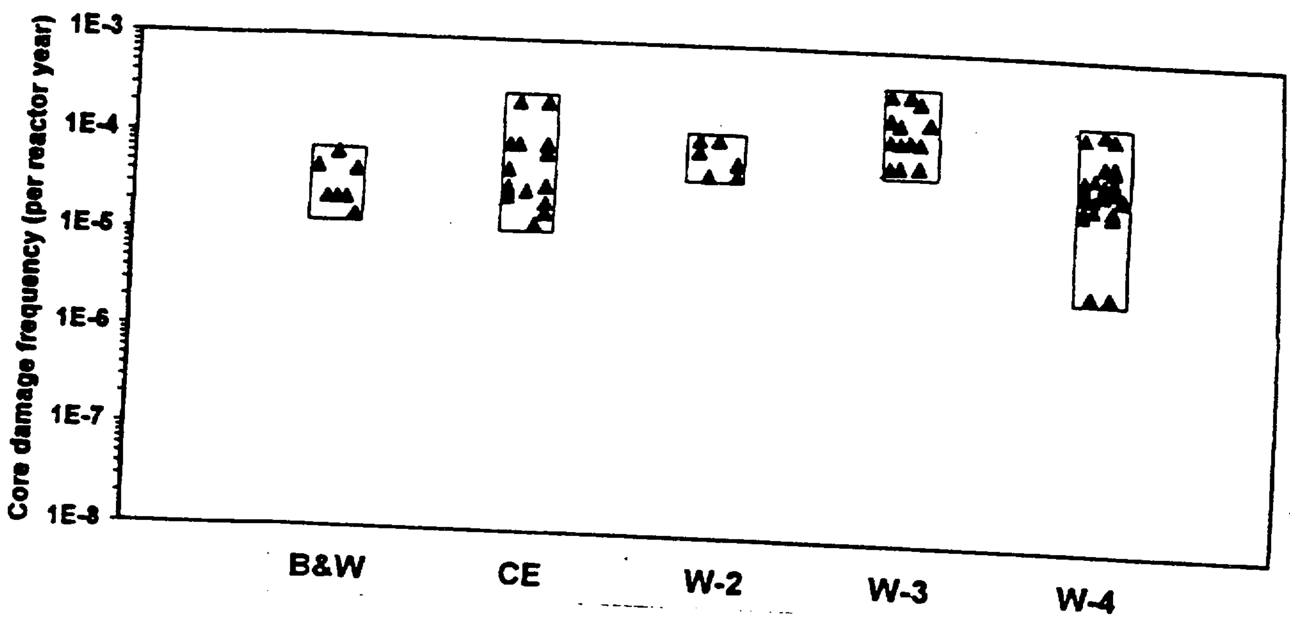
Probability density of estimated external-events core damage frequency for the Surry PWRs



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

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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

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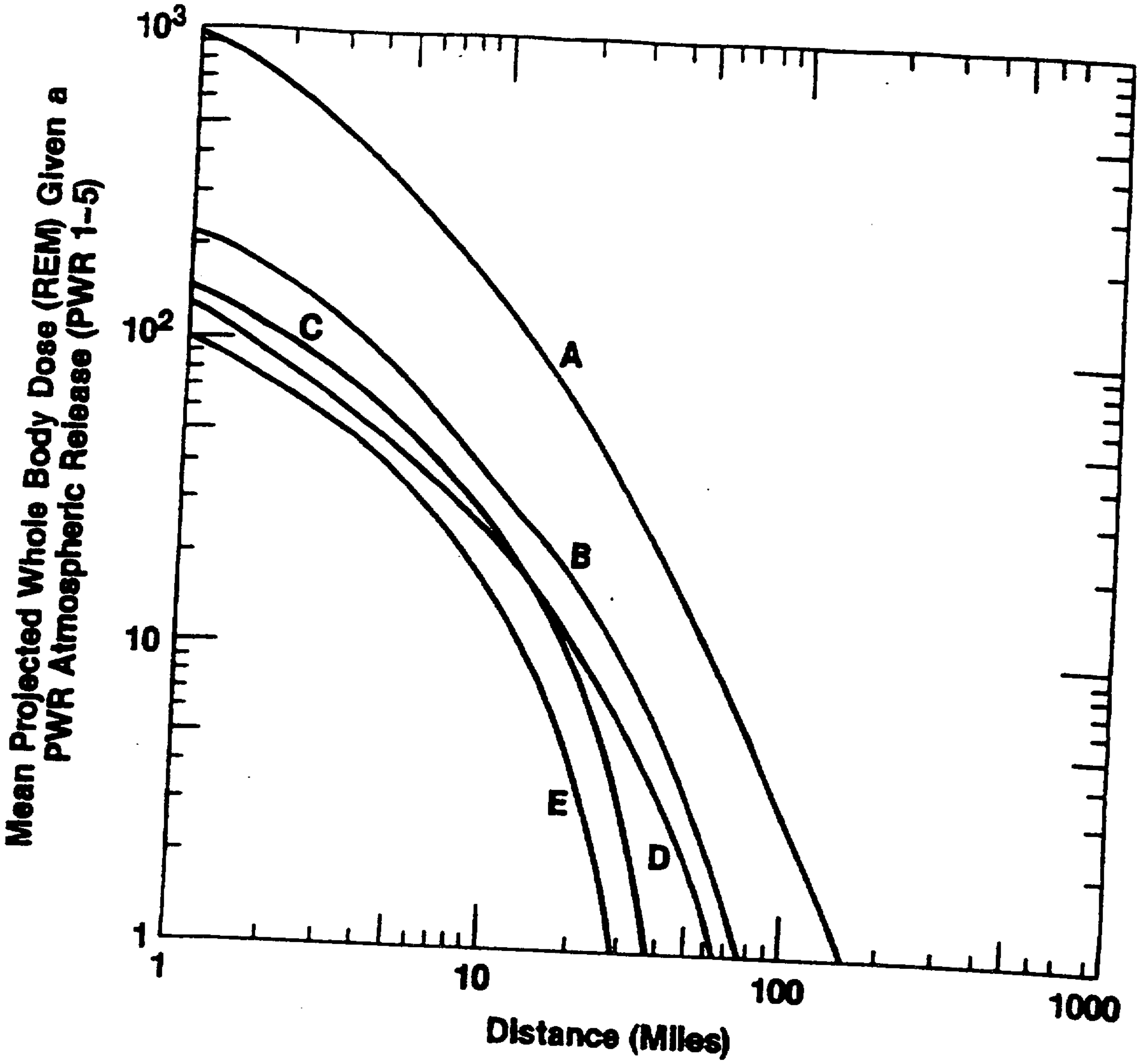
[9] B Flowers et al, Royal Commission on Environmental Pollution, Sixth Report, Cmnd. 6618, Her Majesty's Stationery Office, London, September 1976, page 23.
[10] License amendment application, Enclosure 7, page 5-8.

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

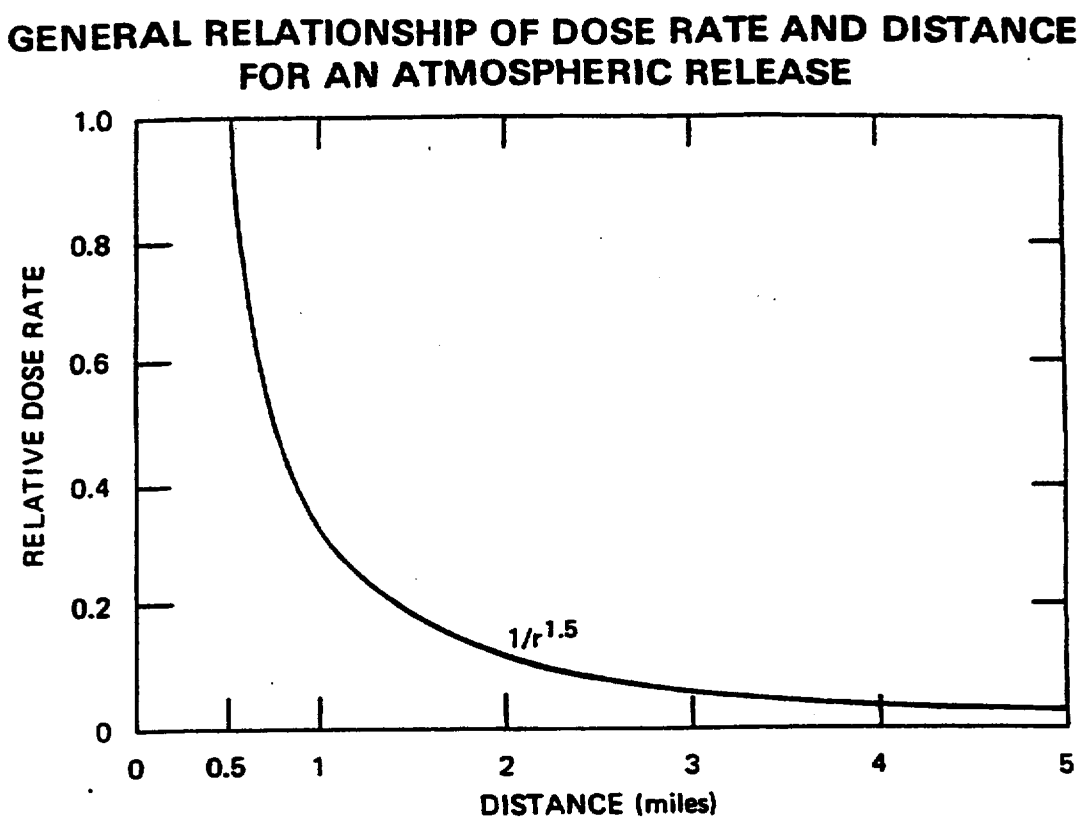


Figure C-2

Dose-distance relationship for a severe reactor accident

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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

[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

[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.

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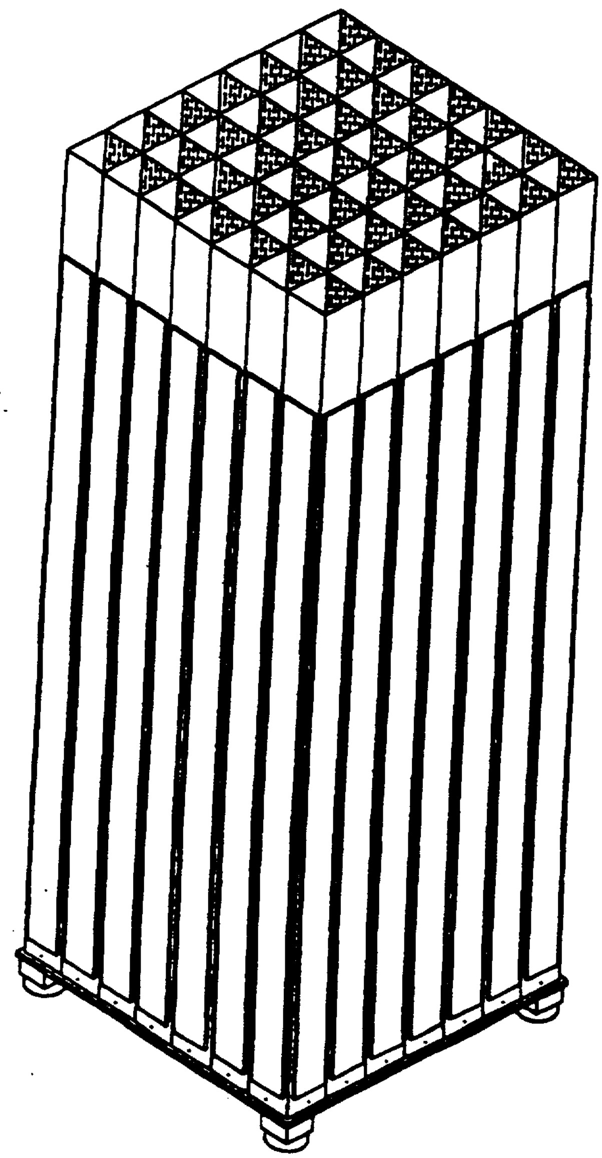
12 N A Pisano et al, The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool, Draft Report, January 1984.
13 V L Sailor et al.
14 Ibid.

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.

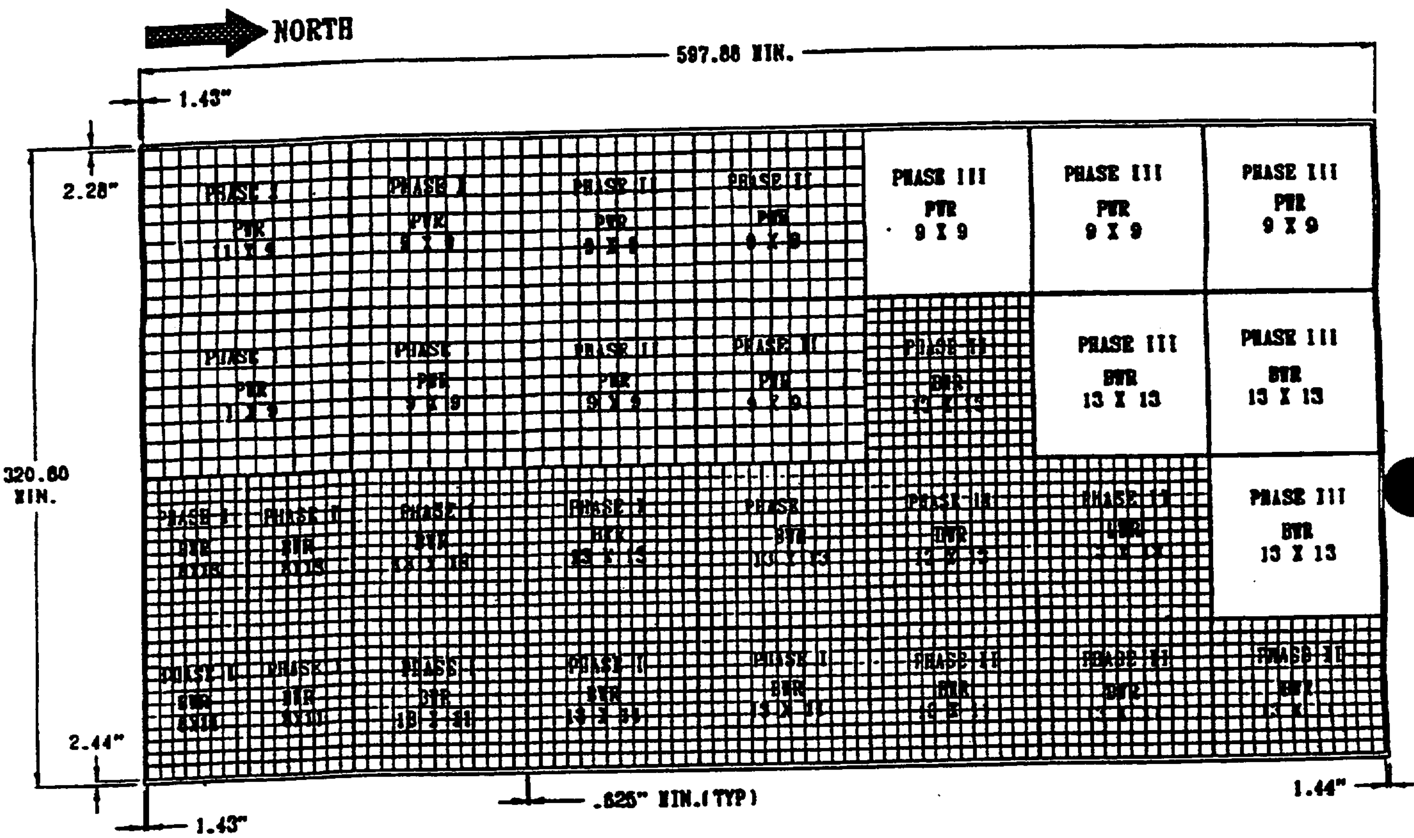
Appendix E considers two potential releases of cesium-137 from the Harris pools. One release corresponds to an exothermic reaction in fuel aged 9 years or less. The other release corresponds to a reaction in fuel aged 3 years or less.



Source: License amendment application

Figure D-1

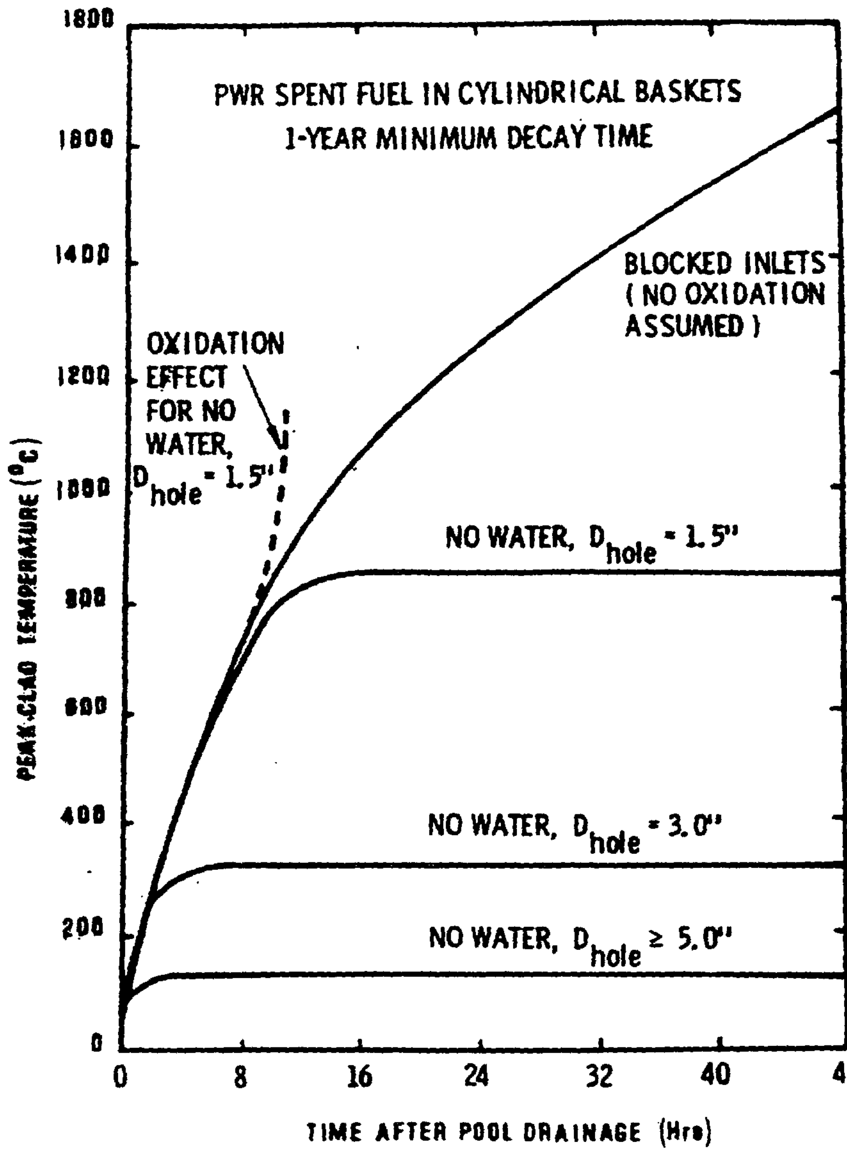
Typical rack used in the Harris pools



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

001776

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

[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

[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.

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

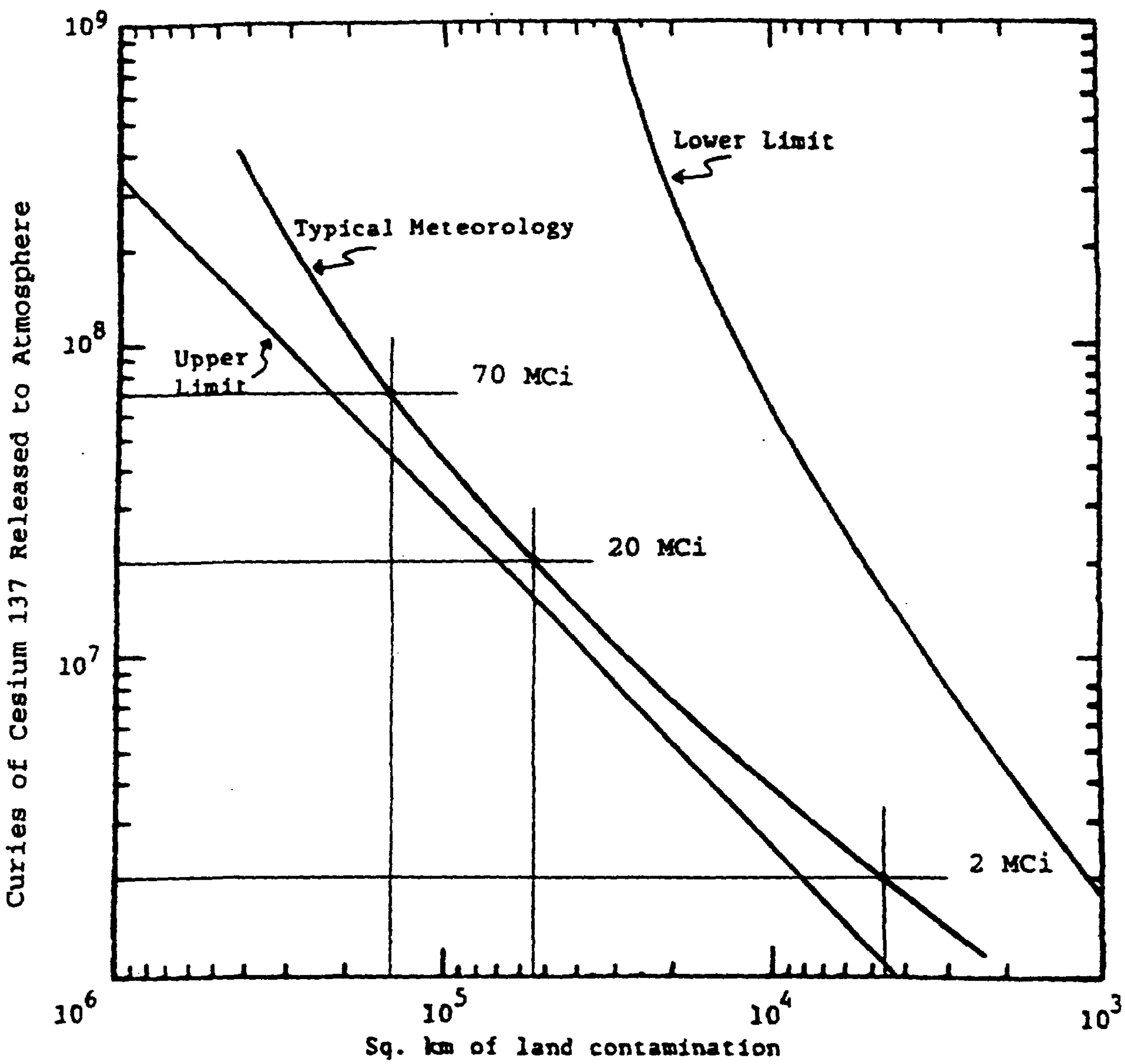

Figure E-1

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

CASE SCHEDULED FOR ORAL ARGUMENT SEPTEMBER 5, 2002

In the
United States Court of Appeals
For the District of Columbia Circuit

Nos. 01-1073 and 01-1246 (Consolidated)

ORANGE COUNTY, NORTH CAROLINA, *Petitioner*
v.
UNITED STATES NUCLEAR REGULATORY COMMISSION
And the UNITED STATES OF AMERICA, *Respondents*
CAROLINA POWER & LIGHT COMPANY, *Intervenor-Respondent*

PETITION TO REVIEW A FINAL DECISION OF THE U.S. NUCLEAR REGULATORY COMMISSION

EXHIBITS

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Dated: June 4, 2002

EXHIBITS

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Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants

October 2000

[pages 2-1 - 2-2 follow]

2.0 THERMAL-HYDRAULIC ANALYSES

Analyses were performed to evaluate the thermal-hydraulic characteristics of spent fuel stored in the spent fuel pools (SFPs) of decommissioning plants and determine the time available for plant operators to take actions to prevent a zirconium fire. These are discussed in Appendix 1A. The focus was the time available before fuel uncovery and the time available before the zirconium ignites after fuel uncovery. These times were utilized in performing the risk assessment discussed in Section 3.

To establish the times available before fuel uncovery, calculations were performed to determine the time to heat the SFP coolant to a point of boiling and then boil the coolant down to 3 feet above the top of the fuel. As can be seen in Table 2.1 below, the time available to take actions before any fuel uncovery is 100 hours or more for an SFP in which pressurized-water reactor (PWR) fuel has decayed at least 60 days.

Table 2.1 Time to Heatup and Boiloff SFP Inventory Down to 3 Feet Above Top of Fuel (60 GWD/MTU)

DECAY TIME	PWR	BWR
60 days	100 hours (>4 days)	145 hours (>6 days)
1 year	195 hours (>8 days)	253 hours (>10 days)
2 years	272 hours (>11 days)	337 hours (>14 days)
5 years	400 hours (>16 days)	459 hours (>19 days)
10 years	476 hours (>19 days)	532 hours (>22 days)

The analyses in Appendix 1A determined that the amount of time available (after complete fuel uncovery) before a zirconium fire depends on various factors, including decay heat rate, fuel burnup, fuel storage configuration, building ventilation rates and air flow paths, and fuel cladding oxidation rates. While the February 2000 study indicated that for the cases analyzed a required decay time of 5 years would preclude a zirconium fire, the revised analyses show that it is not feasible, without numerous constraints, to define a generic decay heat level (and therefore decay time) beyond which a zirconium fire is not physically possible. Heat removal is very sensitive to these constraints, and two of these constraints, fuel assembly geometry and spent fuel pool rack configuration, are plant specific. Both are also subject to unpredictable changes as a result of the severe seismic, cask drop, and possibly other dynamic events which could rapidly drain the pool. Therefore, since the decay heat source remains nonnegligible for many years and since configurations that ensure sufficient air flow[2] for cooling cannot be assured, a zirconium

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[2]Although a reduced air flow condition could reduce the oxygen levels to a point where a fire would not be possible, there is sufficient uncertainty in the available data as to when this level would be reached and if it could be maintained. It is not possible to predict when a zirconium fire would not occur because of a lack of oxygen. Blockage of the air flow around the fuel could be

fire cannot be precluded, although the likelihood may be reduced by accident management measures.

Figure 2.1 plots the heatup time air-cooled PWR and BWR fuel take to heat up from 30 °C to 900 °C versus time since reactor shutdown. The figure shows that after 4 years, PWR fuel could reach the point of fission product release in about 24 hours. Figure 2.2 shows the timing of the event by comparing the air-cooled calculations to an adiabatic heatup calculation for PWR fuel with a burnup of 60 GWD/MTU. The figure indicates an unrealistic result that until 2 years have passed the air-cooled heatup rates are faster than the adiabatic heatup rates. This is because the air-cooled case includes heat addition from oxidation while the adiabatic case does not. In the early years after shutdown, the additional heat source from oxidation at higher temperatures is high enough to offset any benefit from air cooling. This result is discussed further in Appendix 1A. The results using obstructed airflow (adiabatic heatup) show that at 5 years after shutdown, the release of fission products may occur approximately 24 hours after the accident.

In summary, 60 days after reactor shutdown for boildown type events, there is considerable time (>100 hours) to take action to preclude a fission product release or zirconium fire before uncovering the top of the fuel. However, if the fuel is uncovered, heatup to the zirconium ignition temperature during the first years after shutdown would take less than 10 hours even with unobstructed air flow. After 5 years, the heatup would take at least 24 hours even with obstructed air flow cases. Therefore, a zirconium fire would still be possible after 5 years for cases involving obstructed air flow and unsuccessful accident management measures. These results and how they affect SFP risk and decommissioning regulations are discussed in Sections 3 and 4 of this study.

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caused by collapsed structures and/or a partial draindown of the SFP coolant or by reconfiguration of the fuel assemblies during a seismic event or heavy load drop. A loss of SFP building ventilation could also preclude or inhibit effective cooling. As discussed in Appendix 1A, air flow blockage without any recovery actions could result in a near-adiabatic fuel heatup and a zirconium fire even after 5 years.

NUREG-1560
Vol. 1

Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance

Part 1

Final Summary Report

Manuscript Completed: October 1997
Date Published: December 1997

Division of Systems Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

000005

Table 1.3 Summary of PWR plant classes and associated nuclear power plants.

Class	IPE submittals
Babcock & Wilcox (B&W)	• Arkansas Nuclear One,1 • Crystal River 3 • Davis-Besse • Oconee 1,2&3 • Three Mile Island 1
	The B&W plants use once-through steam generators. Primary system feed-and-bleed cooling can be established through the pressurizer power relief valves using the high-pressure injection (HPI) system. The HPI pump shutoff head is greater than the pressurizer safety relief valve setpoint. Emergency core cooling recirculation (ECCR) requires manual alignment to the containment sumps. The reactor coolant pumps (RCPs) are generally a Byron Jackson design.
Combustion Engineering (CE)	• Arkansas Nuclear One,2 • Calvert Cliffs 1&2 • Fort Calhoun 1 • Maine Yankee • Millstone 2 • Palisades • Palo Verde 1,2&3 • San Onofre 2&3 • St. Lucie 1&2 • Waterford 3
	The CE plants use U-tube steam generators with mixed capability to establish feed-and-bleed cooling. Several CE plants are designed without pressurizer power-operated valves. The RCPs are a Byron Jackson design.
Westinghouse 2-loop	• Ginna • Kewaunee • Point Beach 1&2 • Prairie Island 1&2
	These plants use U-tube steam generators and are designed with air-operated pressurizer relief valves. Two independent sources of high-pressure cooling are available to the RCP seals. Decay heat can be removed from the primary system using feed-and-bleed cooling. ECCR requires manual switchover to the containment sumps. The RCPs are a Westinghouse design.
Westinghouse 3-loop	• Beaver Valley 1 • Beaver Valley 2 • Farley 1&2 • North Anna 1&2 • Robinson 2 • Shearon Harris 1 • Summer • Surry 1&2 • Turkey Point 3&4
	This group is similar in design to the Westinghouse 2-loop group. The RCPs are a Westinghouse design.
Westinghouse 4-loop	• Braidwood 1&2 • Byron 1&2 • Callaway • Catawba 1&2 • Comanche Peak 1&2 • DC Cook 1&2 • Diablo Canyon 1&2 • Haddam Neck • Indian Point 2 • Indian Point 3 • McGuire 1&2 • Millstone 3 • Salem 1&2 • Seabrook • Sequoyah 1&2 • South Texas 1&2 • Vogtle 1&2 • Watts Bar 1 • Wolf Creek • Zion 1&2
	The Westinghouse 4-loop group includes nine plants housed within ice condenser containments. Many of these plants have large refueling water storage tanks such that switchover to ECCR either is not needed during the assumed mission time or is significantly delayed. The RCPs are a Westinghouse design.

000006

Generic Environmental Impact Statement for License Renewal of Nuclear Plants (NUREG-1437 Vol. 1)

[SECTION 6.4.6.3 FOLLOWS]

Availability Notice

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000007

disposal capacity would probably not be needed before about the year 2040 to avoid storing spent fuel at a reactor for more than 30 years after expiration of reactor operating licenses.

Second, the NWPA prohibits the opening of an MRS until a permanent repository has been selected and constructed (Pub. L. 97-425). Moreover, the findings of environmental assessments for the MRS and permanent repository must be incorporated in facility design (DOE/RW-0187; GAO/RCED-90-103). Both of these requirements could cause additional delays in the availability of an MRS or permanent repository, necessitating longer on-site storage of the additional spent fuel. Current efforts to identify a host site for an MRS are unlikely to provide for a completed facility by 1998 (GAO/RCED-91-194).

Third, plant refurbishment during license renewal may also adversely affect spent-fuel storage capacity. Utilities may use fuel pools for interim storage of reactor components, as is being done at Vermont Yankee.

During the license renewal period, utilities will focus increasingly on dry storage methods for spent fuel. Either wet or dry storage would meet NRC's Waste Management Confidence Decision Review (49 FR 171; 10 CFR 50 and 51; 54 FR 187), but dry storage is growing in favor because it is more stable. Enlarging spent-fuel racks, adding racks to existing pool arrays, reconfiguring spent fuel with neutron-absorbing racks, and employing double-tiered storage will continue to be pursued; however, above-ground dry storage, utility sharing of spent fuel, and increased fuel burnup to reduce spent-fuel volumes will be the most favored methods until a permanent off-site repository or MRS becomes available, as shown by the study sample and industry-wide survey (Roberts 1987; Mullen et al. 1988; Zacha 1988; Johnson 1989; Fisher 1988).

Industry experience with spent-fuel storage, coupled with supplemental studies of the integrity of pool and dry storage systems, indicates that spent fuel generally can be stored safely on site with minimal environmental impacts (55 FR 38474; NUREG-1092). However, a maintenance concern with spent-fuel pools at permanently closed power plants was identified recently (*Nuclear Waste News* 1994). In January 1994, at the permanently shutdown (since 1978) Dresden Unit 1, a large amount of pool water leaked from a frozen service-water pipe located in the unheated containment building. Because the spent fuel had cooled for 15 years, lowering the pool water depth in this case did not cause significant increases in worker exposure. However, this incident has led to additional safety precautions' being implemented at all permanently shutdown plants.

Extended pool storage provides a benign environment that does not lead to degradation of the integrity of spent-fuel rods. Moreover, continuing advances in dry storage techniques, particularly in standardization of procedures and equipment, indicate that these systems are simple, passive, and easily maintained (53 FR 31651; NUREG-1092; Mullen et al. 1988).

For pool storage, while plant life extension could possibly increase the likelihood of inadvertent criticality through dense-racking or spent-fuel handling accidents, NRC regulations are in place to satisfactorily address this problem. In addition, studies of fuel rod or cladding failures indicate that fuel rods should remain secure well beyond the period of plant life extension, if it becomes necessary to continue pool storage on site (EPRI NP-3765; AIF/NESP-032; EPRI NP-5983; Bailey 1990; Gilbert et al. 1990; 55 FR 38474).

As a result of the operational experience demonstrated by Surry, Robinson, Oconee, and Ft. St. Vrain, NRC has determined that ISFSI methods of dry storage are sufficiently well developed, safe, and dependable to permit the generic licensing for any nuclear plant licensee (provided the plant licensee notifies NRC of the intent to use an ISFSI, uses NRC-certified casks, follows all specified conditions for their use, and provides a full description and safety assessment of the proposed site for an ISFSI) (55 FR 29181; 53 FR 31651). Worker and population exposures are minimal, and ISFSIs use only a small fraction of available land. Environmental assessments undertaken for all ISFSIs have resulted in issuance of findings of no significant impact (NRC Dockets 72-2, 72-3, 72-4, and 72-9).

The principal occupational exposures from spent-fuel management will occur during repackaging of spent-fuel rods and during construction and handling activities associated with moving and storing spent-fuel bundles and racks. While these impacts are expected to vary by the amount of fuel requiring storage, occupational doses during the period of license renewal are not expected to result in doses in excess of present levels (Section 4.6.3). Environmental impacts to on-site land availability should be minimal, given the small amount of land required for expanded spent-fuel pools and dry storage facilities.

6.4.6.3 On-Site Storage of Spent Fuel

Current and potential environmental impacts from spent-fuel storage have been studied extensively and are well understood. Storage of spent fuel in spent-fuel pools was considered for each plant in the safety and environmental reviews at the construction permit and operating license stage. The Commission has studied the safety and environmental effects of the temporary storage of spent fuel after cessation of reactor operation and published a generic determination of no significant environmental impact in its regulations at 10 CFR 51.23. The

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environmental impacts of storing spent fuel on site in a fuel pool for 10 years prior to shipping for off-site disposal were assessed and are included within the environmental data given by Table S-3, found in the Commission's regulations at 10 CFR 51.51. Environmental assessments (EA) for expanding the fuel-pool storage capacity have been conducted for more than 50 plants. A finding of no significant environmental impact was reached for each fuel-pool capacity expansion. Dry cask storage at an ISFSI is the other technology used for spent-fuel storage on site. The Commission has conducted EAs for seven site-specific licensed ISFSIs and has reached a finding of no significant environmental impact for each. The Commission has recently amended its regulations in 10 CFR 72 to allow power reactor licensees to store spent fuel on their sites under a general license. The environmental impacts of implementing this rule were analyzed in an EA that incorporated EAs performed for previous rulemakings related to 10 CFR 72 and for the Commission's Waste Confidence Decision.

At the construction permit and operating license stage, both the 10 CFR 50 safety review and the 10 CFR 51 environmental review contributed to understanding the potential radiological and nonradiological environmental impacts of fuel-pool construction and operation. The design and operating conditions of spent-fuel pools and their various auxiliary systems were reviewed to ensure that the design criteria of Appendix A to 10 CFR 50 are met. These criteria address (1) control of releases of radioactive materials to the environment, (2) fuel storage and handling and radioactivity control, (3) prevention of criticality in fuel storage and handling, (4) monitoring fuel and waste storage, and (5) monitoring radioactive releases. These criteria ensure that radioactive releases to the environment are controlled and acceptable and that effluent discharge paths and the plant environs are monitored for radioactivity. Appendix I to 10 CFR 50 provides the numerical objectives for the design objectives and limiting conditions for operation required to meet the ALARA criterion for radioactive material in the total effluent from an LWR. The objectives were quoted earlier in this chapter and include an objective that total radioactive material in liquid effluent should not result in an annual dose or dose commitment to the total body or to any organ of an individual in an unrestricted area for all pathways of exposure in excess of 5 mrem. In addition, the calculated annual total quantity of radioactive material, except tritium and dissolved gases, should not exceed 5 Ci for each reactor at a site. Appendix I objectives for annual total gaseous effluent of radioactive material for all reactors at a site is that gamma radiation doses should not exceed 10 mrad and beta radiation doses should not exceed 20 mrad for an individual located at or beyond the site boundary. Radioactive materials from the spent-fuel pool contribute a small fraction of the total radioactive materials released from a plant. It is the total releases that need to meet Appendix I numerical objectives. In the construction permit and operating license review for each plant, a thorough assessment is made of calculated releases of curies per year of radioactive materials in both liquid effluent and in gaseous effluent, the exposure pathways, and the impacts to man and biota other than man.

The Commission has considered whether radioactive wastes generated in nuclear power reactors can be subsequently disposed of without undue risk to the public health and safety and the environment. As stated in its regulations at 10 CFR 51.23:

(a) The Commission has made a generic determination that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impact for at least 30 years beyond the licensed life for operation (which may include the term of a revised or renewed license) of that reactor at its spent-fuel storage basin or at either on-site or off-site independent fuel storage installations. Further, the Commission believes that there is reasonable assurance that at least one mined geological repository will be available within the first quarter of the twenty-first century, and sufficient repository capacity will be available within 30 years beyond the licensed life for operation of any reactor to dispose of the commercial high-level waste and spent fuel originating in such reactor and generated up to that time.

In accordance with this determination the rule also provides that no discussion is required concerning environmental impacts of spent-fuel storage for the period following the term of the reactor operating license, including a renewed license. The waste confidence determination was first published in 1984 at 49 FR 34694, August 31, 1984 and was amended in 1990 at 55 FR 38474, September 18, 1990. Additional information and explanation of the safety and environmental considerations supporting the waste confidence determination are given in the notice of the proposed rule amendment, 54 FR 39767, September 28, 1989.

The environmental impacts of storing spent fuel on site in a fuel pool for 10 years prior to shipping for off-site disposal are incorporated in the data presented in Table S-3. The environmental impacts of storage of spent fuel in a fuel pool are given in Table 2.5 of NUREG-0116, *Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle*. Commitment of land, water consumption, chemical effluent, gaseous, liquid and solid radiological effluent, and thermal effluent are all negligible.

Since 1984, licensees have continued to provide safe and environmentally innocuous additional reactor-pool storage capacity through reracking. Over 50 reviews for the expansion of fuel-pool capacity have been completed by the Commission. Each review has resulted in a finding of no significant environmental impact. The reracking activities take place within existing structures and already disturbed land areas, and the changes in radiological, nonradiological, and thermal effluent are all negligible.

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Dry storage of spent fuel at ISFSI has been extensively studied by the Commission, and the environmental impacts are well understood. Licensing requirements for the independent storage of spent fuel and HLW are given in 10 CFR 72. In part, these regulations cover siting evaluation factors, general design criteria, general license for storage of spent fuel at power reactor sites, and approval of spent-fuel storage casks.

6.4.6.4 On-Site Dry Cask Storage

On-site dry cask storage of spent fuel can be accomplished either by a specific license issued under 10 CFR 72.40 or by the provisions of a general license issued under 10 CFR 72.210 for an ISFSI at operating power reactors. To date, seven specific licenses have been issued under 10 CFR 72.40 and one general license issued under 10 CFR 72.210 is operational. For each specific license the Commission has prepared an EA and a finding of no significant impact. Each EA addressed the impacts of construction, use, and decommissioning, including fugitive dust, erosion, noise, heat, and radiological impacts. The Commission also prepared an EA for the general license issued on July 18, 1990 (55 FR 29191). The Commission does not prepare an EA for each general licensee but does prepare an EA for each dry storage cask listed under 10 CFR 72.214 which is approved for use by general ensees. Currently seven casks are listed under 10 CFR 27.214 and it is anticipated that more will be added. General licensees can use only casks listed under 10 CFR 72.214.

EAs prepared for site-specific licenses include site description, need for action, alternatives, site and environment, description of the ISFSI, environmental impacts of proposed action, safeguards for spent fuel, decommissioning, and finding of no significant impact. Under the environmental impacts of the action, the following are considered: land use and terrestrial resources, water use and aquatic resources, noise and air-quality impacts of construction, socioeconomic impacts of construction, radiological impacts of construction, radiological impacts of routine operations, off-site dose, collective occupational dose, radiological impacts of off-normal events and accidents, land use and terrestrial resources, water use and aquatic resources, other effects of operation, and resources committed.

Using the Calvert Cliffs Nuclear Power Plant Site ISFSI EA as typical, the following impacts are evaluated. Land use is about six acres, which is within the owner-controlled area of 2300 acres. During construction of the pad, water for cleaning, drinking, and fugitive dust control was transported to the site by truck. Storm-water runoff and sediment were controlled according to local codes. Air quality had a temporary increase of suspended particulate material, hydrocarbons, carbon monoxide, and oxides of nitrogen from construction activities. The size of the work force was not expected to exceed 50 people. This expanded work force had little impact in the area with large population growth. During initial construction there were no radiological impacts. As construction proceeded, after filling some storage modules, radiation was controlled with temporary shielding to meet NRC and ALARA exposure requirements. Dry storage of spent fuel in welded canisters has no gaseous or liquid effluents. The exposure of the nearest resident, 4705 ft from the facility, when the facility is filled with design-basis spent fuel in 120 modules, the license limit, is less than one mrem/year. The exposure of that resident from other operations at the site is 13.5 mrem/year. These exposures are well within the requirements of 10 CFR 72.104 and 40 CFR Part 190 limits of 25 mrem/year. By year 2010 there are projected to be about 500 people living between 1 and 2 miles of the Calvert Cliffs Station. The collective dose is estimated to be about 101 man-rem/year. Occupational exposure in constructing additional modules after the initial set has been loaded is expected to total about 4 man-rem. Once all 120 modules are loaded, the radiation exposure from the ISFSI is expected to be less than 5 percent of the total site yearly exposure of 350 man-rem. Worst-case accident dose was calculated to be 23 mrem to the whole body and 111 mrem to the thyroid at the nearest residence. Heat from the modules is not expected to be high enough to affect vegetation growth. Fences will discourage some wildlife species from using the area adjacent to the modules. There is no planned use of water or liquid discharge to local surface or groundwater supplies. Surface runoff from precipitation will enter the Chesapeake Bay under existing drainage routes, but it is not expected to result in negative impact to water quality. Rain may vaporize and form a localized fog over the modules that would not extend beyond the plant exclusion boundary. Noise during construction and movement of fuel would not be distinguishable from other operational noise at the site or to result in adverse impact to local residents. The Commission believes that the impacts discussed above reasonably describe the impacts from existing dry cask storage facilities, as well as the likely impacts from those dry cask storage facilities that are expected to be constructed as a result of license renewal.

The Commission prepares an EA for each approved cask listed in 10 CFR 72.214. These EAs are tiered off the "Final Waste Confidence Decision," August 31, 1984 (49 FR 34688), the *Environment Assessment for 10 CFR 72 "Requirements for the Independent Storage of Spent Fuel and High-Level Radioactive Waste,"* NUREG-1092 (August 1984), and the "Environmental Assessment for Proposed Rule Entitled `Storage of Spent Nuclear Fuel in NRC-Approved Storage Casks at Nuclear Power Reactor Sites,'" for the proposed rule published on May 5, 1989 (54 FR 19379). Additional impacts evaluated are those associated with the construction, use, and disposal of the cask. These impacts are very small compared to the total impact of the steel industry, plastics industry, and the concrete industry. The incremental impacts of cask use are considered small. No effluents, either gaseous or liquid, are expected from the sealed casks. Incremental radiation doses off site are also considered to be small compared to those from the other operations on the site. Based on the above summary a finding of no significant initial ISFSI license or amendment for which application is made is required in any environmental report, environmental impact statement, environmental assessment or other analysis prepared in connection with certain actions. This rule affects only the licensing and operation of nuclear power plants. Entities seeking or holding Commission licenses for such facilities do not fall within the scope of the definition of small businesses found in section 34 of the Small Business Act, 15 U.S.C. 632, in the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR part 121, or in the NRC's size standards published December 9, 1985 (50 FR 50241).

Backfit Analysis

This final rule does not modify or add to systems, structures, components or design of a facility; the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility. Accordingly, no backfit analysis pursuant to 10 CFR 50.109(c) is required for this final rule.

List of Subjects in 10 CFR Part 51

Administration practice and procedure, Environmental impact statement, Nuclear materials, Nuclear power plants and reactors, Reporting and recordkeeping requirements.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 552 and 553, the NRC is adopting the following amendment to 10 CFR part 51.

PART 51—ENVIRONMENTAL PROTECTION REGULATIONS FOR DOMESTIC LICENSING AND RELATED REGULATORY FUNCTIONS

1. The authority citation for part 51 continues to read as follows:

Authority: Sec. 161, 68 Stat. 948, as amended (42 U.S.C. 2201); secs. 201, as amended, 202, 88 Stat. 1242, as amended, 1244 (42 U.S.C. 5841, 5842).

Subpart A also issued under National Environmental Policy Act of 1969, secs. 102, 104, 105, 83 Stat. 853-854, as amended (42 U.S.C. 4332, 4334, 4335); and Pub. L. 95-604, Title II, 92 Stat. 3033-3041. Sections 51.20, 51.30, 51.60, 51.61, 51.80, and 51.97 also issued under secs. 135, 141, Pub. L. 97-425, 96 Stat. 2232, 2241, and sec. 148, Pub. L. 100-203, 101 Stat. 1330-223 (42 U.S.C. 10155, 10161, 10168). Section 51.22 also issued under sec. 274, 73 Stat. 688, as amended by 92 Stat. 3036-3038 (42 U.S.C. 2021) and under Nuclear Waste Policy Act of 1982, sec. 121, 96 Stat. 2228 (42 U.S.C. 10141). Sections 51.43, 51.67, and 51.109 also issued under Nuclear Waste Policy Act of 1982, sec. 114(f), 96 Stat. 2216, as amended (42 U.S.C. 10134(f)).

2. Section 51.23, paragraph (a) is revised to read as follows:

§ 51.23 Temporary storage of spent fuel after cessation of reactor operation—generic determination of no significant environmental impact.

(a) The Commission has made a generic determination that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the licensed life for operation (which may include the term of a revised or renewed license) of that reactor at its spent fuel storage basin or at either onsite or offsite independent spent fuel storage installations. Further, the Commission believes there is reasonable assurance that at least one mined geologic repository will be available within the first quarter of the twenty-first century, and sufficient repository capacity will be available within 30 years beyond the licensed life for operation of any reactor to dispose of the commercial high-level waste and spent fuel originating in such reactor and generated up to that time.

* * * * *

Dated at Rockville, Maryland this 11th day of September, 1990.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 90-21889 Filed 9-17-90; 8:45 a.m.]

BILLING CODE 7590-01-D

10 CFR Part 51

Waste Confidence Decision Review

AGENCY: Nuclear Regulatory Commission.

ACTION: Review and Final Revision of Waste Confidence Decision.

SUMMARY: On August 31, 1984, the Nuclear Regulatory Commission (NRC) issued a final decision on what has come to be known as its "Waste Confidence Proceeding." The purpose of that proceeding was "...to assess generically the degree of assurance now available that radioactive waste can be safely disposed of, to determine when such disposal or offsite storage will be available and to determine whether radioactive waste can be safely stored onsite past the expiration of existing facility licenses until offsite disposal or storage is available." (49 FR 34658). The Commission noted in 1984 that its Waste Confidence Decision was unavoidably in the nature of a prediction, and committed to review its conclusions "...should significant and pertinent unexpected events occur or at least every five years until a repository is available." The purpose of this notice is to present the findings of the Commission's first review of that Decision.

The Commission has reviewed its five findings and the rationale for them in light of developments since 1984. This revised Waste Confidence Decision supplements those 1984 findings and the environmental analysis supporting them. The Commission is revising the second and fourth findings in the Waste Confidence Decision as follows:

Finding 2: The Commission finds reasonable assurance that at least one mined geologic repository will be available within the first quarter of the twenty-first century, and that sufficient repository capacity will be available within 30 years beyond the licensed life for operation (which may include the term of a revised or renewed license) of any reactor to dispose of the commercial high-level radioactive waste and spent fuel originating in such reactor and generated up to that time.

Finding 4: The Commission finds reasonable assurance that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the licensed life for operation (which may include the term of a revised or renewed license) of that reactor at its spent fuel storage basin, or at either onsite or offsite independent spent fuel storage installations.

The Commission is reaffirming the remaining findings. Each finding, any revisions, and the reasons for revising or reaffirming them are set forth in the body of the review below.

The Commission also issued two companion rulemaking amendments at the time it issued the 1984 Waste Confidence Decision. The Commission's reactor licensing rule, 10 CFR part 50, was amended to require each licensed reactor operator to submit, no later than five years before expiration of the operating license, plans for managing spent fuel at the reactor site until the spent fuel is transferred to the Department of Energy (DOE) for disposal under the Nuclear Waste Policy Act of 1982 (NWPA). 10 CFR part 51, the rule defining NRC's responsibilities under the National Environmental Policy Act (NEPA), was amended to provide that, in connection with the issuance or amendment of a reactor operating license or initial license for an independent spent fuel storage installation, no discussion of any

[PAGES 38,475-38,479 omitted]

The standard contracts between DOE and generators of spent nuclear fuel or persons holding title to spent fuel currently provide that in return for payment to the Nuclear Waste Fund, DOE will dispose of high-level waste and spent fuel beginning no later than January 31, 1998. The Commission believes it would be inappropriate for NRC to take any position on the need for generators and those holding title to such material to provide interim storage for it beyond 1998. This is a matter that will have to be resolved between the parties to the standard contracts. NRC, in its original Waste Confidence Decision and in the Proposed Waste Confidence Decision Review, addressed the issue of storage of spent fuel until a repository becomes available and has expressed its confidence that spent fuel will be safely managed until a repository is available. Furthermore, in its original Waste Confidence Proceeding, NRC amended its reactor licensing rule, 10 CFR part 50 to require each licensed reactor operator to submit, no later than five years before expiration of the operating license, plans for managing spent fuel at the reactor site until the spent fuel is transferred to DOE for disposal.

In the Nuclear Waste Policy Act (NWPA), Congress placed primary responsibility for interim storage of spent fuel on the nuclear utilities until disposal becomes available. Section 132 of the NWPA requires that DOE, NRC, and other authorized Federal officials take such actions as they believe are necessary to encourage and expedite the effective use of available storage, and necessary additional storage, at the site of each civilian nuclear power reactor.

Sections 218(a) and 133 of the NWPA also provide that NRC by rule establish procedures for the licensing of any technology approved by NRC for use at the site of any civilian nuclear power reactor. NRC may by rule approve one or more dry spent fuel storage technologies for use at the sites of civilian power reactors without, to the maximum extent practicable, the need for additional site-specific approvals. Congress is eminently aware of the likely need for at-reactor storage of spent fuel and has taken legislative action with respect to this matter. Therefore, the NRC believes it is not necessary to inform Congress of this need. However, the NRC will continue to exercise its responsibility to assure that spent fuel is managed safely until a repository is available and will notify Congress of any actions it believes are necessary to provide this assurance.

2.4 The Commission's Fourth Finding

The Commission finds reasonable assurance that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the licensed life for operation (which may include the term of a revised or renewed license) of that reactor at its spent fuel storage basin, or at either onsite or offsite independent spent fuel storage installations.

Issue No. 13: Consideration of the Cumulative Impacts on Waste Management in the NRC's NEPA Documentation

Comment

DOE commented that the cumulative impacts on waste management of potential reactor operating license extensions should be considered in the NRC's National Environmental Policy Act (NEPA) documentation for license renewals.

NRC Response

DOE has observed that renewal of operating licenses would increase the total amount of spent fuel requiring disposal or interim storage which would be taken into account in DOE program planning and should be considered in NRC's NEPA documentation for license renewals. This is generally consistent with the discussion in the Commission's proposed decision, especially 54 FR 39795 (third column). The greater amount of spent fuel which must be stored as a result of license renewal does not affect the Commission's overall finding of no significant environmental impacts.

Issue No. 14: Need for NRC to Facilitate ISFSI License Extensions to Reflect the Commission's Revised Fourth Finding

Comment

The Virginia Electric & Power Company (VEPCo) states that the current license on the Independent Spent Fuel Storage Installation (ISFSI) for its Surry nuclear power plant expires on July 31, 2006. VEPCo states that the NRC should initiate actions to facilitate ISFSI license extensions to reflect the proposed revised Fourth Finding that spent fuel generated in any reactor can be safely stored for at least 30 years beyond the licensed life for operation of that reactor either onsite or offsite.

NRC Response

The Commission's Waste Confidence finding on the duration of safe storage of spent fuel is generic in nature. Site-specific licensing procedures remain effective. Pursuant to § 72.42, an ISFSI license is issued for a period of 20 years but may be renewed upon application by the licensee. Part 72 in no way precludes licensees from requesting additional extensions of license terms for ISFSIs. The licensee thus has the option of requesting an ISFSI license renewal to coincide with whatever operating term and post-operation spent fuel storage period is in effect for a particular reactor. For example, a single renewal could extend the Surry ISFSI license expiration date to the year 2026. The NRC does not believe that further revisions to § 72.42 to facilitate these license extensions are warranted at this time.

Issue No. 15: Insufficient Assurance on Duration of Safe Storage and Risk of Fire at a Spent Fuel Pool

Comment

Public Citizen stated that there is not adequate assurance that spent fuel will be stored safely at reactor sites for up to 30 years beyond the expiration of reactor operating licenses. This is even more the case if license extensions of up to 30 years are included. Public Citizen further stated that "the (Waste Confidence) policy statement fails to recognize that spent fuel buildup at reactor sites poses a growing safety hazard. The pools are not well protected from the environment (in many cases they are outside the reactor's containment structure) and have leaked in the past. For example, in December 1986 at the Hatch nuclear power plant in Baxley, Georgia, 141,000 gallons of radioactive water leaked out of the plant's fuel pool. More than 80,000 gallons of the water drained into a swamp and from there into the Altamaha River near the plant." Public Citizen added that "More recently, on August 16, 1988, a seal on a fuel pool pump failed at the Turkey Point nuclear plant near Miami, FL, causing some 3,000 gallons of radioactive water to leak into a nearby storm sewer. The shoes and clothing of approximately 15 workers were contaminated."

Public Citizen also stated that the danger posed by an accident in which enough pool water escaped to uncover the irradiated fuel assemblies would be greater than the operational incidents described above. According to the commenter, if a leak or pump failure caused the water level in a spent fuel pool to drop to a level which exposed the fuel assemblies, the remaining water might be insufficient to provide adequate cooling. The pool water could then heat to the boiling point, producing steam and causing more water to boil away. The danger then is that heat could continue to build up even further until the cladding which encloses the irradiated fuel pellets catches fire. The commenter continued saying that the

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NRC itself, in the time since the original Waste Confidence Decision, has studied the issue of storage in reracked spent fuel pools and concluded in a 1987 report that the consequence of such a cladding fire could be a "significant" radiation release. The NRC report found:

(1) the natural air flow permitted by high-density storage racks is so restricted that potential for self-sustaining cladding fire exists; and

(2) with high-density racks providing "severely restricted air flow" the oxidation (burning) would be "very vigorous" and "failure of both the fuel rods and the fuel rod racks is expected."

Public Citizen states that nowhere in the Proposed Waste Confidence Decision Review does the NRC take into account the findings of this report, which should have been included.

NRC Respons

The Commission has addressed the safety of extended post-operational spent fuel storage at considerable length in the discussion of its proposed revised Fourth Finding.

Operational occurrences cited in Public Citizen's comment have been addressed by the NRC staff at the plants listed. The NRC has taken inspection and enforcement actions to reduce the potential for such operational occurrences in the future. We would like to note, however, that the event at the Hatch plant occurred in a transfer canal between spent fuel pools during an operation that would not normally be performed following expiration of a reactor operating license. In the case of the event at Turkey Point, the water that flowed outside the building went back into the intake of the plant cooling canal. The canal is a large, closed loop onsite flow path. There was no radiation release offsite, and the safety significance of the event appears to have been very low.

Regarding the risk of fire at spent fuel pools, the NRC staff has spent several years studying in detail catastrophic loss of reactor spent fuel pool water possibly resulting in a fuel fire in a dry pool. The 1987 report, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82" (NUREG/CR-4982), referred to in Public Citizen's comment represents an early part of the NRC's study. Its findings were based on generic data on seismic hazards and response of spent fuel pools, which resulted in calculated risk numbers with wide ranges of uncertainty. (See p. xiii.) Subsequent study of the consequences and risks due to a loss of coolant water from spent fuel pools was conducted by the NRC, and the results were published in NUREG/CR-5176, "Seismic Failure and Cask Drop Analysis of the Spent Fuel Pools at Two Representative Nuclear Power Plants," January 1989, and NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools," April 1989. These reports were cited in the Commission's Proposed Waste Confidence Decision Review (54 FR 39767-39797, at p.39795, September 28, 1989). Also issued in 1989, as part of the NRC staff's study, was "Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools" (NUREG/CR-5281).

The analyses reported in these studies indicate that the dominant accident sequence which contributes to risk in a spent fuel pool is gross structural failure of the pool due to seismic events. Risks due to other accident scenarios (such as pneumatic seal failures, inadvertent drainage, loss of cooling or make-up water, and structural failures due to missiles, aircraft crashes and heavy load drops) are at least an order of magnitude smaller. For this study, older nuclear power plants were selected, since the older plants are more vulnerable to seismic-induced failures.

It should be noted that for a zircaloy cladding fire in a spent fuel storage pool, an earthquake or other event causing a major loss of cooling water would have to occur within two years after operation of a PWR or six months after operation of a BWR. (See NUREG-1353, p. 4-11.) Thus, during the decades of post-operational storage, even a major loss of cooling water would not be sufficient to cause a cladding fire. During the time the pool would be most vulnerable to a fire, the most-recently discharged fuel assemblies would have to be adjacent to other recently discharged assemblies for a fire to propagate to the older fuel. Considering that a third of the reactor core is typically unloaded as spent fuel each year, the probability of a fire involving even the equivalent of a reactor core--a small portion of a pool's capacity--is quite remote.

It should also be noted that even if the timing of a spent fuel pool failure were conducive to fire, a fire could occur only with a relatively sudden and substantial loss of coolant--a loss great enough to uncover all or most of the fuel, damaging enough to admit enough air from outside the pool to keep a large fire going, and sudden enough to deny the operators time to restore the pool to a safe condition. Such a severe loss of cooling water is likely to result only from an earthquake well beyond the conservatively estimated earthquake for which reactors are designed. Earthquakes of that magnitude are extremely rare.

The plant-specific studies following the 1987 generic study found that, because of the large safety margins inherent in the design and construction of their spent fuel pools, even the more vulnerable older reactors could safely withstand earthquakes several times more severe than their design basis earthquake. Factoring in the annual probability of such beyond-design-basis earthquakes, the plant-specific and generic followup studies calculated that the average annual probability of a major spent fuel pool failure at an operating reactor was ten to thirty times lower than the average probabilities in the 1987 study. (See NUREG/CR-5176, p. xiii, and NUREG-1353, pp. ES-2-3.) For either BWR or PWR designs, this probability was calculated at two chances in a million per year of reactor operation. (See NUREG-1353, pp. ES-3-4.)

After evaluating several regulatory options for reducing the risk of spent fuel pool fires, the NRC regulatory analysis concluded that "[t]he risk[s] due to beyond design basis accidents in spent fuel pools, while not negligible, are sufficiently low that the added costs involved with further risk reductions are not warranted." (See NUREG-1353, pp. ES-6-8.)

Issue No. 16: Need for NRC Requirement for Dry Cask Storage Instead of Storage in Spent Fuel Pools

Comment

Public Citizen states that the use of dry cask storage for spent fuel would help address some of the concerns described above, but that NRC has no plans to require dry cask storage instead of storage in spent fuel pools. The commenter notes that NRC has explicitly stated in its Proposed Decision Review that storage in a reactor's "spent fuel storage basin" is considered safe, and (the commenter) apparently disagrees with this conclusion.

NRC Response

The record of operational experience with reactor spent fuel storage pools, as discussed in the Commission's Proposed Decision Review and in response to the preceding comments, strongly supports the conclusion that reactor spent fuel pool storage, which has continued for decades, is safe. Accordingly, the NRC has reached the conclusion that past experience and available information amply support the safety of spent fuel storage, both in pools and dry storage casks, for at least 30 years past the expiration of reactor operating licenses (including the term of a revised license).

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Testimony to the Minnesota Energy Agency, State of Minnesota, Concerning the Proposed Increase of Spent Fuel Storage Capacity at Prairie Island Nuclear Plant

by Gordon Thompson,
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Testimony submitted 10 May 1980
and cross-examined before a
Hearing Examiner of the MEA on
25 June 1980, in Minneapolis.

Testimony to : The Minnesota Energy Agency, State of Minnesota

By : Gordon R Thompson PhD

Concerning : The Proposed Increase of Spent Fuel Storage Capacity at Prairie Island Nuclear Plant

10 May 1980

1. Description of Witness

I am a consultant engineer active in the area of energy and environmental studies and am a member of the Political Ecology Research Group Ltd (a non-profit company) of Oxford, England.

At present I am a consultant to the Center for Energy and Environmental Studies at Princeton University.

The testimony herewith is entirely my own responsibility.

I have previously participated in two major public investigations of the hazards of spent fuel storage, as follows :

(i) In 1977 I prepared and submitted evidence to the Windscale Public Inquiry in UK, on behalf of the Political Ecology Research Group. This evidence addressed the hazards of a proposed expansion of the Windscale reprocessing plant, including the hazards of expanded spent fuel storage.

(ii) During 1978-79 I participated in the Gorleben International Review, a process whereby a group of critical scientists, commissioned by the government of Lower Saxony, reviewed plans for a proposed nuclear fuel center at Gorleben, West Germany. My work for this review included a study of the hazards of spent fuel storage.

2. Nature of this Testimony

This testimony addresses one of the potential hazards of an expanded storage of spent fuel at the Prairie Island plant in the manner proposed by Northern States Power Company.

The potential hazard addressed is that of a loss-of-coolant accident affecting the spent fuel pools at Prairie Island, leading to a release to the atmosphere of radioactive material.

3. Cooling of the Spent Fuel under Normal Conditions

The plan of Northern States Power Co is to cool the expanded holding of spent fuel assemblies by natural circulation of water, horizontally beneath the base-plate of each spent fuel rack and vertically upwards through the storage tubes within which the fuel assemblies are confined. The pool water is then to be cooled by heat exchangers, the heat ultimately being discharged to cooling towers and the Mississippi River.

This plan differs from the present practice at Prairie Island by virtue of the higher density of fuel assemblies. That higher density demands that each fuel assembly be surrounded by a tube made of stainless steel and neutron absorbing material. The presence of this tube means that coolant (ie water) can reach each fuel assembly only via the base of its tube.

4. Potential Circumstances Leading to Loss-of-Coolant

There are essentially two ways in which coolant (ie water) could be lost :

- by evaporation
- by breach of a pool

Loss by Evaporation

If the operation of the pool-water cooling system were interrupted, the water would, after some hours, begin to boil. If no water were added to the pool, then evaporation would eventually reduce the water level sufficiently that fuel assemblies would be exposed to the air.

To appreciate the time-scale for this process, consider the reference case for accident circumstances as outlined in Appendix A. That case is at the more severe end of the spectrum of possible accident circumstances, as regards heat production from the spent fuel and inventory of radioactive material in the pool.

- Appendix B outlines the calculations which show, for the Appendix A reference case, the following progression of events :

Cooling of pool-water ceases :	t = 0 hrs
Water begins to boil :	t = 20 hrs
Sufficient water has boiled away so that 1/2 of length of fuel assemblies is exposed to air :	t = 135 hrs

The obvious question is : "Under what circumstances could this situation arise ?"

To answer : The most probable circumstances are those associated with a reactor accident. At Prairie Island the spent fuel pools are located immediately adjacent to the twin reactor containment buildings and the pools share many systems with the reactors (cooling, water-makeup and control systems). Thus a severe reactor accident is likely to interfere with the normal operation of the pools.

A severe reactor accident could be associated in many different ways with fire or explosion in the containment or auxiliary buildings and/or release of radiation from the containment building. Such radiation release, even if it were not at the worst end of the possible spectrum in regard to contamination of the general environment, could be severe enough to prevent access to the spent fuel pools or their support systems.

Figure 1 illustrates this possibility. Shown there is estimated radiation dose-rate inside a typical PWR containment building for a "design-base" accident, namely one in which the containment building "successfully" confines the radiation. The Salem FSAR, from which this figure is taken, acknowledges that radiation levels in parts of the auxiliary building could be up to 1% of that inside the containment (eg 620 rad/hr after 100 hrs for Prairie Island plant(1)). It will be noted that death within 10-30 days due to bone marrow damage can be expected for persons exposed to radiation in the range of 300-1000 rads(2). Noting also that one certainly cannot exclude a reactor accident which leads to a more severe radiation environment than does the "design-base" accident, it is clear that prevention of access for substantially more than 100 hrs is plausible.

Loss by Breach of a Pool

From Appendix A we see that the reinforced concrete pool walls vary in thickness from 3 to 6 ft. Such walls could be breached by :

- sabotage
- aircraft crash
- earthquake

Of particular importance in the case of Prairie Island is the above-grade location of the pools, as shown in Figure 2. For this arrangement, a breached pool will drain freely. Other reactor pools (eg at Zion plant) are arranged so that the top of the spent fuel is at grade level and so that at least part of the pool walls are surrounded by earth. Consequently, such pools are less at risk regarding rapid drainage than are the Prairie Island pools.

5. Events in a Pool Following Loss-of-Coolant

Initial Heatup of Spent Fuel Assemblies

This process is discussed in Appendix C, from which it will be seen that exposure to air of about 1/2 of the length of the fuel assemblies would lead to fuel cladding temperature in excess of 1000°C.

It is important to note that partial loss of water would lead to higher cladding temperature than would pertain for total water loss.

Reaction of Zircaloy Cladding with Steam

At temperatures above 1000°C, zirconium reacts exothermically with steam, producing hydrogen gas (as occurred during the Three Mile Island accident).

Appendix D discusses this reaction and shows that the reaction, once initiated, would proceed rapidly. A large fraction of the pools' inventory of zirconium could be consumed within 1/2 hr.

Release of Radioactive Material from Spent Fuel Pellets

As outlined in Appendix E, a zirconium-steam reaction would yield heat sufficient that a substantial fraction of the mass of the spent fuel pellets would be melted. In consequence, substantial radioactive release would occur to the atmosphere within the pool building.

Also, as mentioned previously, hydrogen gas would be produced. It should be expected that this accumulation of hydrogen would lead to an explosion which would breach the pool building. In that way, most of the radioactive release estimated in Appendix E would enter the outside atmosphere.

6. Consequences of Atmospheric Release

A full estimate of the health effects and other impacts of such a release would require substantial effort. One would investigate the outcome of various strategies of evacuation, administration of thyroid-blocking medication and interdiction of food supplies.

Some indication of the impact of release can be gained from Figure 3, which shows[3] the area which would be contaminated by differing releases of Cesium 137. It can be seen that the release estimated in Appendix E would contaminate, for typical meteorological conditions, 10,000 - 50,000 km^2 of land. Such an event would be a major catastrophe.

7. Implications of this Hazard Potential

In this context, one can learn from the process of the Gorleben International Review (GIR). Dr Albrecht, governor of the West German state of Lower Saxony, and several of his cabinet, attended a semi-public examination, during 28 March - 2 April 1979, of the contentions of the members of the GIR. This led to a statement[4] by Albrecht on 16 May 1979, containing the following stipulations regarding spent fuel storage :

> "This radioactive potential is so immense that it must not be possible to release it by an incident.
>
> The State Government is not willing to license the concept of DWK in its present form. They insist that the entry store for spent fuel elements is made inherently safe such that the cooling does not depend on the functioning of technical equipment or on human reliability."

The fulfilment of Albrecht's stipulations at Prairie Island would require :

- the construction of an entirely new spent fuel store
- design of the new store to be such that loss-of-coolant would leave cladding temperature below the ignition point
- the quantity of fuel in existing pools, and its density of packing, to be such that loss-of-coolant in those pools would leave cladding temperature below the ignition point

8. Notes

(1) From Figure 1, the Salem dose-rate inside containment is 1.3×10^5 rad/hr. For the Prairie Island plant, we adjust by the ratio (0.48) of the capacity of each Prairie Island reactor (530 MWe) to that of each Salem reactor (1100 MWe), yielding 6.2×10^4 rad/hr in containment and up to 6.2×10^2 rad/hr in the auxiliary building.

(2) H Smith and J W Stather, report NRPB-R52 of UK National Radiological Protection Board, November 1976.

(3) This figure is taken from the report prepared by Jan Beyea (then at the Center for Energy and Environmental Studies, Princeton University) as his contribution to the Gorleben International Review, February 1979.

(4) Chapter 3 ("Potential Accidents and their Effects") of the GIR report can be obtained (in English) from : Political Ecology Research Group, PO Box 14, Oxford, UK. This document includes Albrecht's statement.

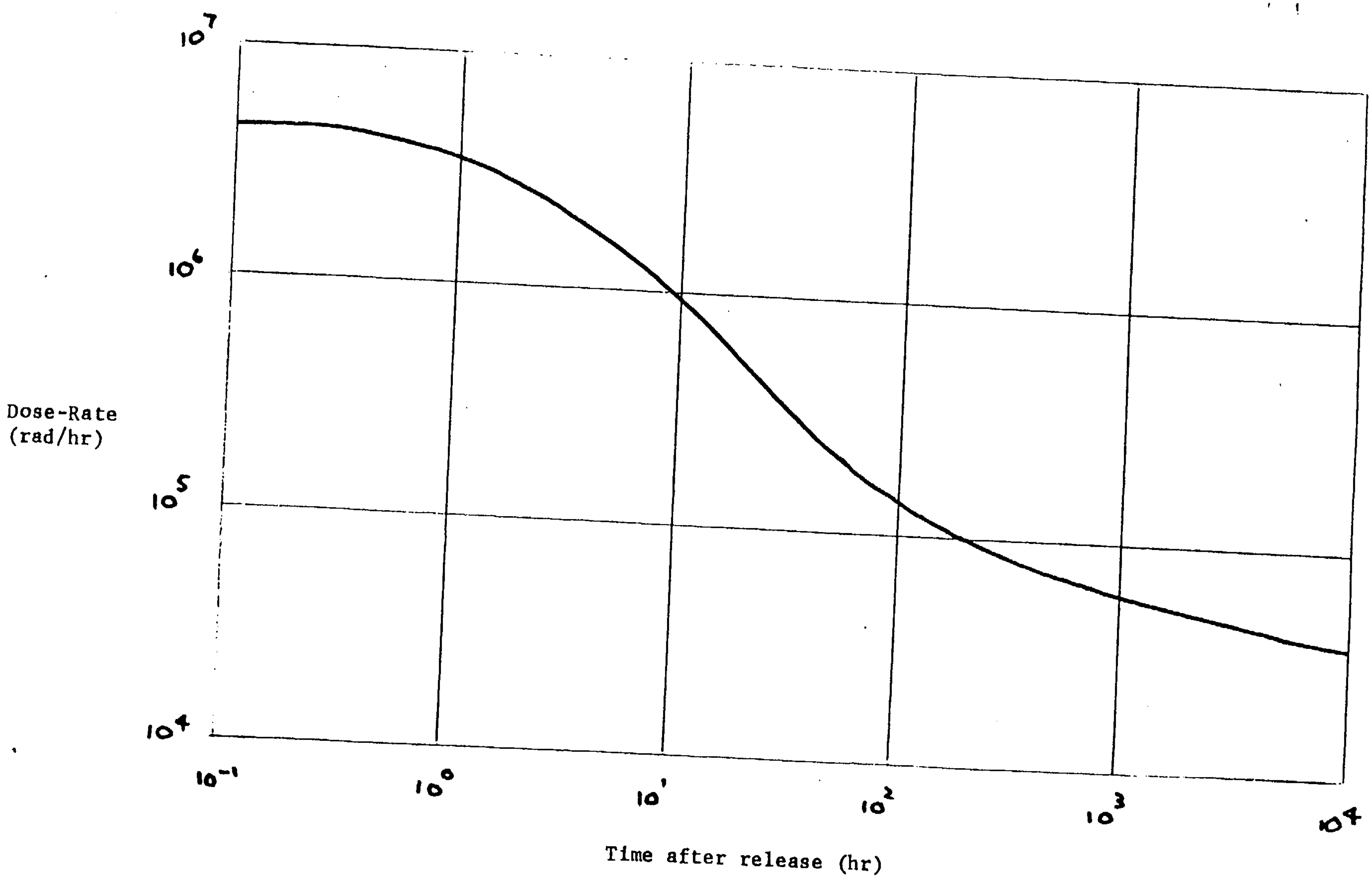

Figure 1: Gamma Dose-Rate Inside Containment Building following Loss-of-Coolant Accident
Source: Final Safety Analysis Report, Salem Units 1 and 2. Public Service Electric and Gas Co.

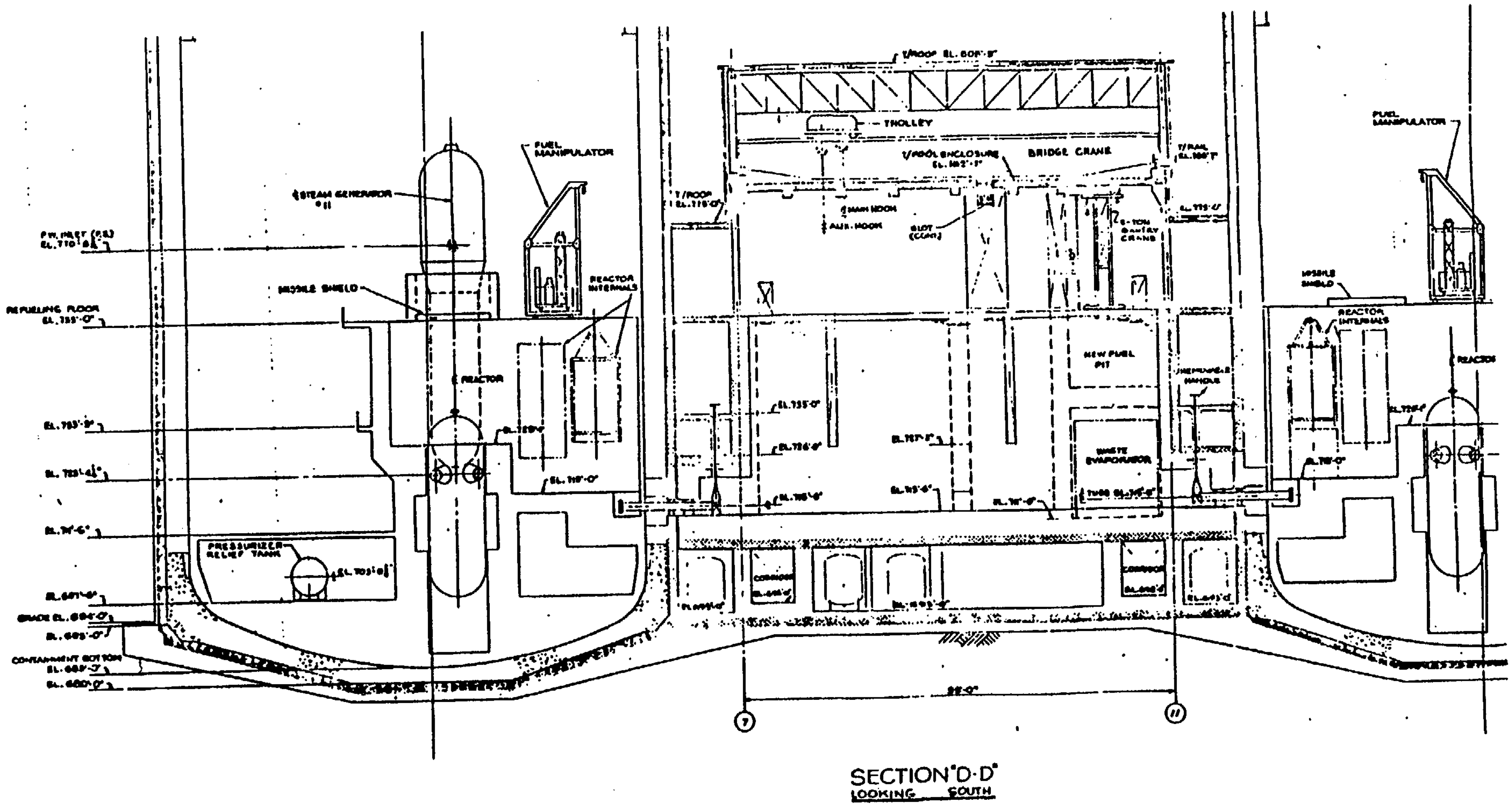

Figure 2 : Cross-Section of Prairie Island Plant Showing Spent Fuel Pools

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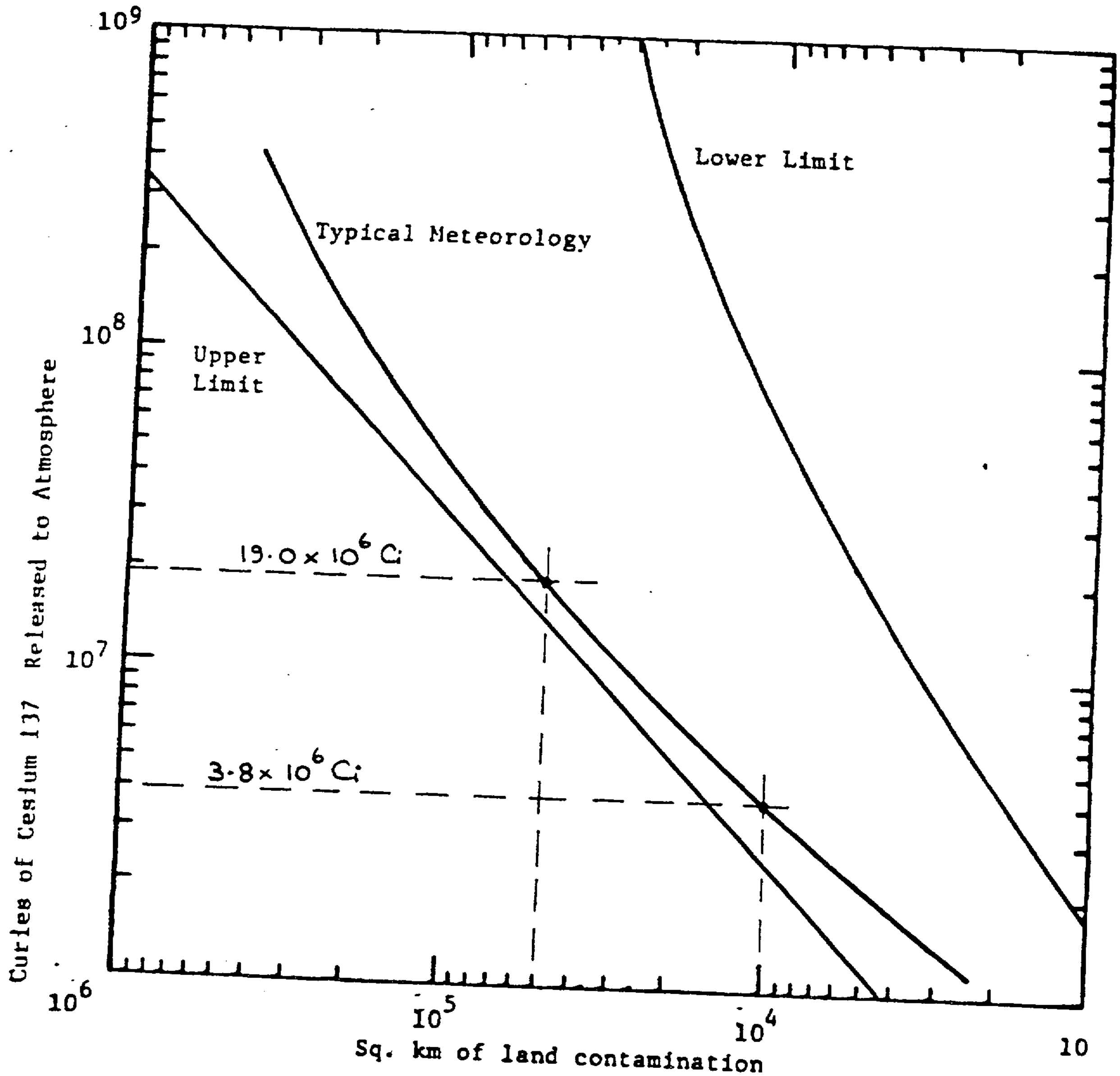

Figure 3 : Area of Land Contaminated by Atmospheric Release of Cesium 137

Notes

(1) The "typical meteorology" curve assumes 5 m/s windspeed, Pasquill stability class D, 0.01 m/s deposition velocity, 1000 m mixing layer and 300 m initial plume rise.

(2) The contamination threshold used is a 10 rem dose in 30 yrs (approx 3 times background).

(3) This figure is taken from a report by Beyea (see note (3) in body of testimony).

Appendix A

Reference Case for Loss-of-coolant Accident

DATA CONCERNING PRAIRIE ISLAND PLANT

(source : Certificate of Need Application submitted to Minnesota Energy Agency by Northern States Power Co, September 1979)

- 2 PWR reactors each of 530 MWe capacity
- 121 fuel assemblies per reactor core
- 40 fuel assemblies removed per refueling
- each fuel assembly contains approx 400 kg of heavy metal
- dimensions of pool 1 are 5.56 m x 5.77 m x 12.29 m (volume 394 m^3)
- dimensions of pool 2 are 13.23 m x 5.77 m x 12.29 m (volume 938 m^3)
- proposed fuel assembly storage tubes are of 8.3 inch inside dimension and 9.5 inch center-to-center spacing
- volume of each fuel assembly is 0.158 m^3
- pool wall thickness is 3-6 ft
- proposed total spent fuel capacity is 1582 assemblies
- normal temperature range of pool water is 105°F to 130°F

REFERENCE CASE

Suppose that one reactor had been refueled 60 days before the accident and that the entire core of the second reactor had been removed 10 days before the accident. Further suppose that the pools contained normal refueling discharge for the previous 15 yrs. The pools' inventory would be :

age of fuel assembly after discharge from reactor	number of fuel assemblies
10 days	120
60 days	40
1 yr	80
.	.
.	.
15 yrs	80
Total :	1360

The characteristics of this spent fuel inventory have been estimated using NRC data (source : NRC report NUREG-0404 , March 1978). It is found that the heat load and inventory of the most important radionuclides would be as follows :

Heat Load : 5.33 MW

(of which 3.84 MW is from the 10-day-old fuel and 0.56 MW is from the 60-day-old fuel)

Inventory of Most Important Radionuclides

Sr 90	2.9×10^7 Ci
Ru 106	3.9×10^7 Ci
I 131	1.9×10^7 Ci
Cs 137	3.8×10^7 Ci
Pu 238	6.7×10^5 Ci

Appendix B

Loss of Pool Water by Evaporation

(data from Appendix A)

The mean boiling temperature of the pools would be 113°C. If the spent fuel heat capacity is assumed to be that of water (volumetrically), and if heat loss to surroundings is neglected, then the time required for the water temperature to rise from its normal level (assumed to be 45°C) to boiling temperature would be 19.8 hrs.

During the boiling phase, the mean latent heat of water would be 2.24 MJ/kg. The fuel assemblies are 4.1 m long (source : replies by Northern States Power Co to questions from the Minnesota Energy Agency, February 1980); thus approx 1/2 of the length of the fuel assemblies would be exposed to air following boil-away of 10 m depth of water. If heat loss to surroundings is neglected, then the additional time required for this would be 114.7 hrs.

Appendix C

Cooling of a Spent Fuel Assembly Partially Exposed to Air

The mechanisms of cooling available to the exposed portion of a fuel assembly are :

- natural convective circulation of air and steam within the fuel storage tube (closed at its bottom end by water)
- conduction along the fuel assembly
- radiation to the pool environment
- superheating, as it rises past the exposed portion of the fuel assembly, of steam generated by the immersed portion of the assembly

The respective heat removal capacities of these mechanisms have been discussed by this author as part of the Gorleben International Review (see note (4) in body of this testimony). It is found that only the last of these mechanisms is significant for fuel cladding temperatures up to several thousand degrees C.

The temperature of superheated steam as it rises past the top of the fuel assembly is, interestingly, independent of the age of the fuel after discharge. It depends only on the fraction of fuel length exposed, as follows :

exposed fraction	maximum steam temperature (^{o}C)
0.3	560
0.4	820
0.5	1180
0.6	1710
0.7	2610

Cladding temperature will of course be greater than steam temperature. It suffices to note that cladding temperature would readily exceed $1000^{o}C$ for an exposed fraction of 0.5 .

- The above comments are confirmed by the results of computer modelling conducted by Sandia Laboratories for the NRC (A S Benjamin et al, "Spent Fuel Heatup Following Loss of Water During Storage", NRC report NUREG/CR-0649, March 1979). It is interesting that the introduction of this report is not consonant with its contents; it states (incorrectly) that "complete drainage" is "the most severe type of spent fuel storage accident".

It should be noted that complete drainage would permit circulation of air beneath the base-plate of the fuel racks and vertically upward through the storage tubes. Partial drainage would block this air circulation.

Appendix D

Reaction of Zirconium with Steam

This reaction is : $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + 6.53$ MJ per kg Zr

(source : p 441, T J Thompson and J G Beckerley (eds), "The Technology of Nuclear Reactor Safety", Vol 2, 1973)

If access of steam is not limited, the reaction rate can be represented by :

$$\frac{da}{dt} = \frac{k}{a} \exp(-C/T)$$

where : a = equivalent thickness of cladding reacted (m)
t = time (sec)
T = cladding temperature (^oK)
C = 22800
k = 3.97×10^{-5}

(source : F C Finlayson, report no 9 of Environmental Quality Laboratory, California Institute of Technology, May 1975)

The 1/a component of this rate law accounts for the inhibiting effect of the growing oxide layer.

For a constant temperature, the time required to completely oxidize the cladding is :

$$\text{Total oxidizing time} = \frac{A^2}{2k} \exp(C/T)$$

where A = total cladding thickness (m)

Typically, $A = 6.2 \times 10^{-4}$ for a PWR, leading to the following results :

cladding temperature (^oC)	total oxidizing time (secs)
1500	1860
2000	110
2500	18

Appendix E

Melting of Spent Fuel Pellets

For the reference case outlined in Appendix A, 617 Mg of UO_2 would be present in the Prairie Island Pools. The ratio of the mass of zircaloy to the mass of UO_2 in a PWR would be 0.207 (source : Reactor Safety Study, WASH-1400, Appendix VIII, 1975); leading to a zirconium inventory in the Prairie Island pools of 128 Mg.

Given a heat of reaction of 6.53 MJ per kg Zr (see Appendix D), complete reaction of the Zr would yield 8.4×10^{11} J .

The heat required to raise the temperature of UO_2 from $300^{o}K$ to just above its melting point ($3030^{o}K$) is 1.2 MJ/kg (source : R A Meyer and B Wolfe, Advances in Nuclear Science and Technology, Vol 4, pp 197-250, 1968); thus the heat required to melt the pools' inventory of UO_2 would be 7.4×10^{11} J .

If there were no heat loss to the surroundings, it is clear that all of the fuel pellets could be melted. A full estimate of the fraction of the mass of the fuel pellets which would actually be melted, and of the release of radioactive material, would require a substantial investigative effort. My preliminary estimate of the release to atmosphere of radionuclides is :

I, Cs, Ru	:	10-50 %
Sr, Pu	:	1 %

This leads to an estimate of release inventory of the most important radionuclides as follows :

Sr 90	:	2.9×10^{5} Ci
Ru 106	:	$(3.9 - 19.5) \times 10^{6}$ Ci
I 131	:	$(1.9 - 9.5) \times 10^{6}$ Ci
Cs 137	:	$(3.8 - 19.0) \times 10^{6}$ Ci
Pu 238	:	6.7×10^{3} Ci

NUREG-0575, *Handling and Storage of Spent Light Water Power Reactor Fuel* (1979)

<u>excerpts</u>:

pages ES-1 - ES-6

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pages 4-9 – 4-22

NUREG-0575

NUREG-057
Executive S
Text

final generic environmental impact statement

on

HANDLING AND STORAGE OF SPENT LIGHT WATER POWER REACTOR FUEL

AUGUST 1979

Project No. M-4

U. S. Nuclear Regulatory Commission • Office of Nuclear M
Safety and Safegua

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EXECUTIVE SUMMARY

1.0 SCOPE

The Generic Environmental Impact Statement on spent fuel storage was prepared by the Nuclear Regulatory Commisson staff in response to a directive from the Commissioners published in the Federal Register, September 16, 1975 (40 FR 42801). The Commission directed the staff to analyze alternatives for the handling and storage of spent light water power reactor fuel with particular emphasis on developing long range policy. Accordingly, the scope of this statement examines alternative methods of spent fuel storage as well as the possible restriction or termination of the generation of spent fuel through nuclear power plant shutdown.

Since the Commission's directive was issued, there have been significant policy developments. In this regard, the President has stated that the U.S. should defer domestic plutonium recycle in order to search for better solutions to the proliferation problem. In light of the President's views and public comments, the NRC terminated on December 23, 1977, its proceedings on the Generic Environmental Statement on Mixed Oxide Fuel (GESMO), pending license applications, and other matters related to the reprocessing and recycle of spent light water reactor fuel. This policy decision highlights the importance of this GEIS.

On October 18, 1977, the Department of Energy (DOE) announced that the Federal Government would accept and take title to spent nuclear fuel from utilities upon payment of one time storage fees. The new policy is designed to meet the needs of nuclear reactors for both interim and permanent disposition of spent fuel. The DOE policy actions presume continued light water reactor power generation with discharge of spent fuel and government responsibility for the storage and disposition of spent fuel. Thus, these policy actions also address the issues examined in this document. However, this document does continue to serve the function of supporting the need for rulemaking for away-from-reactor (AFR) spent fuel storage facilities. In addition, DOE used this NRC statement as a source in their draft generic environmental impact statement on their announced spent fuel policy.

The storage of spent fuel addressed in this generic environmental impact statement is considered to be an interim action, not a final solution. The Commission has clearly distinguished between permanent disposal and interim storage.[1] Nonetheless, it has expressed its concern that storage of spent fuel not be used to justify retarding the development of a practicable method of permanent disposal.[2] This concern is shared by groups who have studied this situation.[3,4] The Commission is initiating a proceeding to review its basis for confidence that safe waste disposal will be available.[5] The Commission announcement of September 16, 1975, outlining this study stipulated that the Staff was to examine the period through the mid-1980's. In the absence of a national policy directed to final disposition of spent fuel, the staff extended the time period of this study to year 2000. This extension provided a conservative upper bound to the interim spent fuel storage situation at a date that constituted a practical limit to the forecasting that may logically be used as a basis for today's decisionmaking.

The study covers the following:

(1) The magnitude of the possible shortage of spent fuel storage capacity.

(2) The options for dealing with the problem, including, but not necessarily limited to:

- Permitting the expansion of spent fuel storage capacity at nuclear power plants;
- Permitting the expansion of spent fuel storage capacity at reprocessing plants;
- Licensing of independent spent fuel storage facilities;
- Storage of spent fuel from one or more reactors at the storage pools of other reactors (transshipment between reactors); and
- Ordering the generation of spent fuel be stopped or restricted (by shutting down reactors).

(3) A cost-benefit analysis of the alternatives listed in (2) above along with other reasonably feasible options, including:

- Impacts on the public health and safety and the common defense and security;
- Environmental, social and economic costs and benefits;
- Commitments of resources;
- Implications regarding options available for the intermediate and long range storage of nuclear waste materials; and
- Relationships between the local short-term uses of the environment and long-term productivity.

(4) The impacts of possible additional transportation of spent fuel that may be required should one or more of the options be adopted;

(5) The need for more definitive regulations and guidance covering the licensing of one or more of the options for dealing with the problem; and

(6) The possible need for amendments to 10 CFR 51.20(e)--the S-3 table which summarizes environmental consideration for the nuclear fuel cycle.

The scope of this study is limited to considerations pertinent to the interim storage of spent fuel. Other issues related to the "back end" of the fuel cycle, such as reprocessing and waste management, are covered elsewhere, e.g., NUREG Reports, 0002 for plutonium recycle (GESMO), 0116 and 0216 for waste management.

2.0 THE POTENTIAL MAGNITUDE OF THE SPENT FUEL STORAGE PROBLEM

The factors which affect the quantity of spent fuel requiring storage in excess of that which can be accommodated at nuclear power plants are:

- The projected generation of spent fuel--which is a function of the growth rate of nuclear power installed capacity, the assumed average annual reactor capacity factor and the reactor fuel management plans.
- The extent to which conventional spent fuel storage pools at nuclear power plants can be modified to increase the spent fuel storage capacity.
- The option of the plant owner to maintain storage reserve capacity to accommodate a full core discharge; and
- The time to develop a means for the permanent disposition of spent fuel by reprocessing or waste management.

2.1 GENERATION OF SPENT FUEL

Generation of spent fuel was projected through the year 2000 (Table ES.1) on the basis of installed reactor generating capacity (in GWe) from NRC data for reactors now operating, under construction and planned, and Energy Information Administration estimates. The staff estimated that 77,000 metric tons of heavy metal (MTHM) as spent fuel will have been discharged by year 2000 and that the total reactor storage capacity in the year 2000 will be 91,000 MTHM if full core reserve (FCR) is not maintained and 77,000 MTHM if FCR is maintained. Total storage capacity values do not indicate capacity restrictions at individual older reactors.

Table ES.1. Projected Generation of Spent Fuel

Year	MTHM-Cumulative*
1980	3,000
1985	13,000
1990	29,000
1995	50,000
2000	77,000

*Does not include ~4700 MTHM of spent fuel discharged prior to 1979 and stored AR and AFR at the end of 1978.

2.2 AT-REACTOR (AR) STORAGE CAPACITY

The spent fuel storage capacity at nuclear power plants has conventionally been designed to accommodate one full core plus one discharge, i.e., about 1-1/3 cores. The rationale was that spent fuel from a given discharge would be shipped offsite for reprocessing before the next annual discharge and capacity would be reserved to accommodate a full core if conditions made it desirable to unload the plant reactor.* However, most pools were equipped with spent fuel

*This capacity is termed full core reserve (FCR).

storage racks which did not fully utilize the available floor space in the pool. In many cases it is now possible to increase at-reactor spent fuel storage capacity by a factor of about 3.0. This compact storage is accomplished by the replacement of existing racks with new racks designed for closer spacing of fuel assemblies and utilizing previously unused floor space. Most nuclear plants have applied to increase their spent fuel storage capacity, and a majority have already received permission to do so.

The maintenance of reserve capacity sufficient to accommodate the full reactor core in the spent fuel storage pool at a nuclear power plant is not a safety matter. However, many power plant owners may consider the maintenance of full core reserve capacity desirable for operational flexibility. Experience has shown that the capacity for fully unloading a reactor has been useful in making modifications and repairs to reactor structural components and for periodic reactor vessel inspections. Such reserve capacity is effectively unused space in the spent fuel storage pool and has the net effect of reducing the available at-reactor spent fuel storage capacity for successive spent fuel discharges.

2.3 REQUIRED AWAY-FROM-REACTOR (AFR) STORAGE

The magnitude of the projected shortfall in AR spent fuel storage capacity equates to the net requirement for away-from-reactor storage at independent spent fuel storage installations (ISFSI). Assuming no curtailment of nuclear power production, the bounding condition used to estimate the required AFR storage capacity is:

- Feasible modifications of power plant pools (compact storage of fuel).

A range or upper bound of AFR storage requirements for this bound may be established by considering (a) no full core storage reserve, and (b) mair nce of a full core reserve (FCR).

The AFR requirements* are summarized for five-year periods for these conditions in Table ES.2 below.

Table ES.2. Away-from-Reactor Spent Fuel Storage Requirements (MTHM)

Year	With Compact Storage	
	Without FCR	With FCR
1980	0	40
1985	730	1,900
1990	3,900	6,300
1995	9,700	14,000
2000	21,000	27,000

*These include the effect of recent reactor basin storage capacity expansion applications for the Oconee Units 1 & 2 basin, for the Big Rock Point Basin and for the Hatch 1 & 2 basins.

3.0 METHODS FOR DEALING WITH THE PROBLEM OF EXTENDED SPENT FUEL STORAGE

3.1 PERMITTING THE EXPANSION OF SPENT FUEL STORAGE CAPACITY AT NUCLEAR POWER PLANTS (COMPACT STORAGE)

In its announcement dated September 16, 1975, the Commission stated its position that, in the public interest, there should be no deferral of individual licensing actions on the expansion of at-reactor spent fuel storage capacity during the period required for the preparation of this assessment. In line with this policy as of January, 1979, applications for modifications to increase storage-pool capacity at 65 operating nuclear power reactors have been received by the NRC. Such modifications have covered both the installation of newer racks with closer spacing of the spent fuel storage positions and the installation of spent fuel storage racks in previously unused spaces.

The actions can be taken without significant effect on public health and safety, and to date 39 of these applications have been approved and actions are proceeding as planned. Each of these applications was evaluated on an individual basis with findings in each case that:

- At-reactor spent fuel storage can be increased,
- The actions can be taken with no sacrifice of public health and safety, and
- The environmental impact of the proposed increased at-reactor spent fuel storage was negligible.

It should be kept in mind that increased at-reactor spent fuel storage involves only aged fuel (at least one year since discharge) which has orders of magnitude less hazard potential than fuel freshly discharged from a reactor (see Sec. 4.2).

3.2 PERMITTING THE EXPANSION OF SPENT FUEL STORAGE CAPACITY AT REPROCESSING PLANTS

There are no reprocessing plants in operation in the United States at the present time. With the NRC decision to terminate the generic study on plutonium recycle use in mixed oxide fuel (GESMO) in December, 1977 [42 FR 65334] in deference to the President's non-proliferation policy, commercial reprocessing has been indefinitely deferred in the United States. The expansion of spent fuel storage at reprocessing plants is technically feasible, but it is not considered a viable alternative for dealing with the problem of spent fuel storage because of the limited potential spaces at the remaining potential reprocessing plant, Allied General Nuclear Services at Barnwell, S.C., which has storage pool capacity for about 400 metric tons.

3.3 LICENSING OF INDEPENDENT SPENT FUEL STORAGE INSTALLATIONS (ISFSI)

This alternative represents the major means of providing interim AFR spent fuel storage.

The former Nuclear Fuel Services, Inc. reprocessing plant is now licensed and operating as an independent spent fuel storage installation. However, NFS has announced its withdrawal from the reprocessing business, and this plant is no longer receiving spent fuel from utilities for extended storage.

The General Electric Company's planned reprocessing plant at Morris, Illinois, has now been declared and licensed as an ISFSI. The initial licensed spent fuel storage capacity of about 100 MTU has been increased to about 750 MTU by installing spent fuel storage racks in its former high level waste storage pool. The plant operation as a "storage only" facility has shown that an independent spent fuel storage installation can be operated with adequate protection of the health and safety of the public.

The Department of Energy testified on January 26, 1979, before the Committee on Interior and Insular Affairs of the House of Representatives that in order to meet its deadline of 1983 for having an operational AFR facility, it is considering the NFS West Valley, the GE Morris, and the AGNS Barnwell facilities to supply storage capacity.

Currently, an increasing interest in independent spent fuel storage installations is being shown by the nuclear power industry. One architect-engineer company has submitted to NRC a standard design of such a facility, to be situated at a reactor site. The NRC staff has reviewed it and issued letters of approval for the design.

The methods of expanding spent fuel storage capacity considered in this assessment show negligible difference in environmental impact and cost with the exception that at-reactor storage pool compact storage is least costly economically, and does not require additional transportation of spent fuel. In view of this, the reference case alternative for expanded spent fuel storage assumes that most additional storage capacity will be provided by AR storage pool compact storage with additional required storage capacity being provided by away-from-reactor (AFR) at ISFSI located either at reactor sites or at separate sites using the available means of wet or dry storage discussed in this statement.

3.4 STORAGE OF SPENT FUEL FROM ONE OR MORE REACTORS AT THE STORAGE POOLS OF OTHER REACTORS (TRANSSHIPMENT)

Temporary relief for the spent fuel storage problem being faced by some of the older nuclear power plants could be alleviated in some cases by shipping spent fuel to newer plants with unused available storage capacity. However, facility operators can be expected to be reluctant to accept spent fuel that may result in prematurely filling their reactor spent fuel storage pools and potentially impacting the supply of electric power to their regions.

Currently, only one application has been approved by the NRC covering this alternative. The staff's analysis shows that intrautility transshipment, when considered in conjunction with compact storage at reactor pools, provides additional relief delaying the need for AFR storage capacity by about three to four years (see Table 3.2), depending upon whether or not full core reserve (FCR) is maintained. The staff also considered the alternative of transshipment in conjunction with compact storage at reactor pools on an unlimited basis with all the nation's reactor pools operating as a single system under a national storage allocation plan. This alternative is not considered feasible under present regulatory conditions; the staff has analyzed it solely as an emergency alternative necessary to ensure continued reactor power generation in the unlikely event that no AFR storage is made available to prevent spent fuel storage capacity shortfalls. Assuming a preemptive federal regulatory authority to allow this alternative to work, unlimited transshipment in theory could delay the need for AFR storage to the late 1990's.

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of this waste production would reach about 50 megatons/yr in year 2000 and its projected growth from 1979-2000 is shown in Appendix C, Figure C.8.

The wastes (gob) produced during benefaction are commonly rich in pyrites (sulfides of iron), trace elements, and heavy metals. The pyrites release sulfuric acid when exposed to normal rock weathering processes, so runoff water from the gob disposal area may be extremely acidic. The runoff water may also carry high concentrations of trace elements and heavy metals. The exact magnitude of the gob volume, acid released, and metals carried in runoff is highly variable and depends on the composition of the coal and benefaction technology employed. Similarly, uranium must pass through milling, enrichment, and fabrication processes. Although uranium milling is analogous to the benefaction of coal, its impacts are more similar to the impacts of milling metals, such as copper. A generic environmental impact statement on uranium milling is now in preparation. The draft statement has been circulated for comment.

Because only a small fraction of the ore is uranium, "the amount of solid tailings is roughly equal to the ore feed rate plus part of the reagents used in the process ...".[16] The tailings may be acidic or alkaline, depending upon the milling process, and will typically be fine particles.

The coal fuel cycle produces ultimate by-products that require ultimate disposal. The burning of coal produces cinders or slag that must be stored temporarily onsite prior to being transported to the ultimate disposal site. The predicted slag production reaches 1.3-3 megatons/yr in year 2000 based on information in Reference 9 and its growth from 1979-2000 is shown in Appendix C, Figure C.9.

Each year the precipitators and scrubbers for a 1,000-MWe plant at 60% capacity could produce 400-650 tons of fly ash and 70-400 kilotons of wet lime-SO_2 residue. The total expected production of collected fly ash and scrubber sludge in year 2000 reaches about 7-12 kilotons/yr and 7-33 megatons/yr respectively and their growth from 1979-2000 is shown in Appendix C, Figures C.10 and C.11. These wastes would require temporary onsite storage (covering as much acreage as the boiler and turbine buildings combined) and then would be transported to some unspecified ultimate disposal site.

4.2 HEALTH IMPACTS

When one examines the human health impacts associated with the alternatives discussed in this environmental impact statement, it appears that there is little incremental impact associated with the reference case spent fuel storage solution. This is due to the relatively inert conditions of spent fuel in storage. Also, increased storage of spent fuel at any facility simply results in the retention of older fuel that would otherwise have gone to reprocessing or disposal. Volatile and non-volatile radionuclides with short half-lives will have decayed to negligible levels. Consequently, the radiological and heat load impacts of this older fuel are factors of ten lower than that of the less cooled fuel and result in a small incremental impact to health and safety. Thus, environmental and health impacts of spent fuel storage are dominated by new spent fuel, and whether older fuel is present or is disposed of has little impact on the health and safety posture as a whole. The principal health impact is associated with incremental radiation dose. This subject is

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treated separately in Section 4.2.1. Section 4.2.5 treats the impacts associated with the termination case alternative of substituting coal fired power generation for nuclear energy.

4.2.1 Reference Case Storage Alternative

4.2.1.1 Normal Operations

The calculated health effects of the nuclear fuel cycle are summarized in Table 4.2.[17] In addition to the indicated potential excess mortality, there could be increases in morbidity due primarily to the incidence of nonfatal cancers.[17] For persons employed by the nuclear industry, the incremental incidence of nonfatal cancers and benign thyroid nodules could possibly be approximately one case per gigawatt-year.[17] For the general public, the incremental increase in morbidity could be about 0.5 case of a nonfatal cancer per gigawatt-year due to the entire nuclear fuel cycle.

Table 4.2. Summary of Excess Mortality Due to Civilian Nuclear Light-Water Reactor Power, per 0.8 Gigawatt-Year Electric

Fuel Cycle Component	Occupational		General Public		Totals
	Accident	Disease	Accident	Disease	
Resource recovery (mining, drilling, etc.)	0.2	0.038	~0	0.085*	0.32
Processing	0.005***	0.042	**	0.026-1.1	0.073-1.1
Power generation	0.01	0.061	0.04	0.016-0.20	0.13-0.3
Fuel storage	**	~0	**	~0	~0
Transportation	~0	~0	0.01	~0	0.01
Reprocessing	**	0.003	**	0.059-0.062	0.057-0.065
Waste Management	**	~0	**	0.001	0.001
Totals	0.22	0.14	0.05	0.18-1.3	0.59-1.7

*These effects indicate that 4060 Ci of ^{222}Rn released from mining the uranium to produce 0.8 GWy(e) would result in 0.085 excess deaths over all time.

**The effects associated with these activities are not known at this time. While such effects are generally believed to be small, they would increase the totals in this column.

***Corrected for factor of 10 error based on referenced value (WASH-1250).

The radiological impact from spent fuel storage is as follows:

- Population dose due to the release of ^{85}Kr from leaking fuel elements
- Occupational exposure of plant personnel incurred while working in the vicinity of the spent fuel storage pool, e.g., changing water purification filters and ion exchange resins.

These types of impacts are generic to spent fuel storage operations regardless of whether such fuel is stored at a nuclear power plant or at an AFR storage facility.

For the "aged" fuel involved in relatively long time storage, ^{85}Kr leakage rates are too low to be detected. However, for the final GESMO, Chapter IV-K, Extended Spent Fuel Storage, a conservative release rate of 1 Ci/MT-year was used. (Based on experience at the GE Morris Operation,[18] this figure could be high by a factor of 10^6). The resultant population dose factors were:

United States = 0.004 man-rem/MT-yr.
Foreign = 0.02 man-rem/MT-yr.

Occupational dose rates, based primarily on at-reactor experience, used in final GESMO were 20 man-rem per 1,000 MT-yr.

The above figures are applicable to conventional water basin storage pools. The figures for the various types of passive dry storage systems under development are expected to be comparable or less. Based on these figures, the calculated doses due to all spent fuel in storage are shown in Table 4.3. Note that the population doses are not corrected for ^{85}Kr decay.

Table 4.3. Radiological Doses from Spent Fuel Storage

Year	MT Fuel in Storage	Occupational Dose Total Body, man-rem	Population Dose, Skin, man-rem U.S.	Foreign
1980	7,600	160	33	150
1985	18,000	360	77	350
1990	33,400	670	140	660
1995	54,300	1,100	230	1,100
2000	81,200	1,600	350	1,600

4.2.1.2 Compact Storage

For the majority of the facilities treated under this alternative, design, construction, and operating data were available. For the rest it was assumed that current practices in these areas would be continued at least through 1986, and that the 1,000-MWe hybrid model power plant as used in GESMO would be used after 1996. Spent fuel is considered stored at the bottom of large pools of filtered, deionized water.

The water serves as a coolant to remove decay heat of the spent fuel, and as a radiation shield for the stored spent fuel. The occupational radiation exposure results from the radioactivity in the water and the required operational activities. The spent fuel contributes a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel.

Radioactivity in the pool water comes from introduction of reactor coolant water into the pool during refueling, the dislodging of crud from the surface of the spent fuel assemblies during handling of the assemblies, and the leakage of fission products from defective spent fuel elements. The rate of introduction of reactor coolant water into the pool with compact storage should not change because the proposed modification does not involve a change in the refueling procedures. Although the proposed modification will increase the total number of assemblies that can be stored in the pool, it is not expected that there would be a significant increase in the number of times the assemblies are handled before shipment offsite. Also, any significant removal of crud from the surface of an assembly would occur during the initial fuel handling when the assembly is transferred from the core to the storage pool. Therefore, there should not be a significant increase in crud introduced to the pool water due to the proposed modification. Experience with spent fuel stored at the GE Morris Plant and at the NFS, New York Plant has indicated that there is little or no leakage of radioactivity from spent fuel which has cooled several months. There should not be a significant increase in leakage activity from spent fuel to the pool because of the proposed modification.

The pool cleanup system serves to clarify and remove the radioactive materials from the pool water. Pool water treatment technology is well developed, and it is not uncommon to find fuel pool water with radioactivity content comparable to the 10 CFR Part 20 limits for occupational uses. Water carried out of the fuel pool by mechanical means or seepage is collected in sumps and recycled through a radwaste cleanup system. Small amounts of pool water eventually reach the environment but only after several levels of radwaste treatment, so that the quantities of radioactivity released are insignificant.

The only gaseous radionuclides released to the atmosphere in significant quantities are the noble gases, principally krypton-85. Some radiation reaches the environment in the form of direct radiation from the fuel within the pool and from the transportation of intermediate level wastes to the final disposal site. Direct radiation in the vicinity of the spent fuel storage pool is extremely low, in the order of one to two millirem per hour. If this were the only contribution to the occupational dose, that dose would be quite small. However, the occupational dose is dominated by the exposures involved in handling and moving the fuel, in handling radwaste, and in decontaminating tools during which time the dose rates are higher. In all other respects, the FCR and no-FCR alternatives proved to have nearly identical radiation impact. However, the additional handling, due to more fuel at the AFR storage involved, in the FCR alternative results in somewhat higher occupational doses than would be true for the no-FCR alternative.

4.2.1.3 "Away-from-Reactor" Storage

At the moment, independent spent fuel storage installations (ISFSI) comprise two licensed fuel pools, the GE installation at Morris, Illinois, and the NFS installation at West Valley, New York, and one facility undergoing licensing, the AGNS facility at Barnwell, South Carolina. These are relatively small facilities with a maximum total capacity of less than 1,000 tonnes. An ISFSI design of about 1100 metric tons pool capacity to be situated at a reactor site and to utilize some reactor facilities, such as electricity, water, and waste processing systems, has been reviewed by the NRC staff.[19] Such an ISFSI, designed to receive spent fuel from several neighboring reactors of a utility, would have reduced transportation (comparable to offsite reactor transshipment) compared to a large regional ISFSI. However, for the purposes of bounding the impacts of this alternative, large ISFSIs with total capacities of the order of 6,000 tonnes in multiple units of about 500 tonnes

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each were assumed.[20] In effect, each independent unit is the size of the currently projected larger fuel pools at reactors and is designed, built, and operated in very much the same manner. Thus, the majority of the radiological impact considerations (including cask handling) are essentially identical. However, in this case, transportation of spent fuel to the facility, assumed to be 1000 miles away, constitutes a major pathway of dose to the environment.[21] The storage of much larger quantities of spent fuel at these facilities would raise the quantities of noble gases released to the atmosphere per storage facility. Also, the much increased fuel load tended to increase the handling dose, thus raising the occupational exposure; while the more specialized design of these facilities resulted in a lowering of radionuclides released to the aquatic environment.

4.2.2 Safety and Accident Considerations

To be a potential radiological hazard to the general public, radioactive materials must be released from a facility and dispersed offsite. For this to happen:

- The radioactive materials involved must be available in a dispersable form,
- There must be a mechanism available for the release of such materials from the facility, and
- There must be a mechanism available for offsite dispersion of such released material.

Although the inventory of radioactive materials contained in 1000 MTHM of aged spent fuels may be in the order of a billion curies or more, very little is available in a dispersable form; there is no mechanism available for the release of radioactive materials in significant quantities from the facility; and the only mechanism available for offsite dispersion is atmospheric dispersion. Increased spent fuel storage with AR or AFR storage normally involves only aged fuel. The underwater storage of aged spent fuels is an operation involving an extremely low risk of a catastrophic release of radioactivity.

The radioactive materials present in a spent fuel storage installation are:

- The spent fuel in storage
- Impurities in the pool water
- The "crud" deposits on the surfaces of the fuel pins and fuel assembly structural components
- Airborne radioactivity, primarily due to entrainment in evaporating pool water
- Impurities removed from the pool waters by filtration and ion exchange treatment
- Wash solutions generated during shipping cask cleanup and miscellaneous decontamination operations
- Dry materials such as contaminated protective clothing, blotting paper, cleaning materials and ventilation system filters.

4.2.2.1 Composition of Spent Fuel

The spent fuel in storage is highly radioactive, with a total inventory of radionuclides in the order of 106 curies per metric ton of contained uranium. The gross radioactivity in curies per metric ton of uranium as a function of time since discharge from a reactor (decay time) is shown in Table 4.4. The decay times were chosen to represent:

Days		Event
0	-	At time of discharge from reactor.
120	-	Typical short storage time of AR spent fuel.
365	-	Nominal decay time for acceptance of spent fuel at an AFR (proposed 10 CFR Part 72).
3,650	-	Time when only long-lived activity remains.

Note that from a gross radioactivity standpoint, the fission product nuclides are predominant throughout the life of spent fuels in storage, but that 96.8% of this activity decays away in the first 120 days and 98.7% is gone in 365 days.

The fission product radionuclides are , emitters, and only those few that enter into biological processes are of major concern. For freshly discharged fuels at a reactor, a principal concern is the 8-day ^{131}I which is absorbed by plants, animals and humans, particularly in natural iodine deficient inland locations. However, since the quantity of ^{131}I present in discharged fuel is reduced by a factor of over a billion times in the first 365 days of decay, it is not a major concern for the storage of spent fuels in an AFR storage facility.

Those fission product nuclides of primary concern under conditions of long term spent fuel storage are ^{85}Kr and ^{134}Cs-^{137}Cs and possibly ^{129}I. These nuclides are present in significant quantities, are soluable in water and biologically mobile. Cesium enters the muscle tissue of animals and man. The isotope ^{129}I has a low specific activity, 1.4 dpm per gram of iodine in the environment where the background radio of ^{129}I to ^{127}I ranges from 4.8×10^{-10} to 3.1×10^{-9}. Thus, to receive a dose of the same order as that natural dose from ^{40}K in the thyroid would require ^{129}I to ^{127}I ratios about 10,000 times background.[22] However, because of its 17-million year half-life, its release to the environment should be minimized.

Table 4.4. Radioactivity Present in Spent Fuels,*
megacuries per metric ton of uranium**

Decay time - days after discharge	0	120	365	3,650
Fission product nuclides***	180	5.84	2.36	0.326
Actinides and their daughter elements***	43.8	0.191	0.167	0.105
Light elements & fuel element construction materials***	0.189	0.046	0.011	0.002

*See Appendix G for tabulation of nuclides present
**Based on metric tons of uranium charged to a reactor
***Source - ORIGEN code - Reference PWR
- Power - 37.5 MW/MTU
- Burnup - 33,000 MWd/MTU
- Plant capacity factor = 80%

Many of the actinides and their daughter elements are also short lived; 99.6% decay away in 120 days. Of those present in aged spent fuel stored in an AFR storage facility, the plutonium isotopes present the most significant potential hazard.

Of the materials of fuel element construction and surface crud deposites, the most significant radionuclide is cobalt-60.

The only way in which the radionuclides in spent fuel could be made available for dispersal is by physical rupturing of fuel pins. As fuel assemblies must be handled under water to provide the necessary protective shielding, a rupture of fuel pins would allow the escape of free gases, primarily ^{85}Kr, and contact of the fuel material by the pool waters. However, as corrosion rates of ceramic fuel materials are low, the only observable effect might be a slight increase in the ^{137}Cs content of the pool waters.

4.2.2.2 Krypton-85

The principal radioactive gas which could escape from defective fuel elements in storage is ^{85}Kr. The evidence to date indicates that the free gases present in fuel pin void spaces leak out rather quickly from defective fuel elements in the reactor and upon discharge, but that the gases which are contained within the fuel pellet matrix have an extremely low diffusion rate and hence a low leak rate. Experience at the NFS West Valley reprocessing plant with chopping fuel, in preparation for dissolution, showed the the release of krypton from spent fuel was marginally observable on their krypton stack monitor; almost all of the krypton was retained in the fuel until its dissolution. This experience indicates that even the rupture of a number of fuel elements in the storage pool would not cause a release of ^{85}Kr in sufficient quantities to be measurable offsite.

4.2.2.3 Cesium-134/137

Stable cesium is rare geologically and in the biosphere but radioactive cesium from weapons testing fallout is widely distributed throughout the biosphere. Cesium-137 is important as it is readily absorbed from the food intake by both animals and man. However, the cesium in spent fuel is strongly bound within the fuel matrix even when the fuel pellets are exposed to the pool water. The dissolution rate of cesium is very low and decreases sharply with time. The cesium concentration in pool waters is readily controllable by circulation through an ion exchange resi bed.

4.2.2.4 Pool Water Activity

The fuel pellets are sintered ceramic cylinders which have a very low solubility in water, and the contained radioactivity is tightly bound within the fuel material. In addition, the fuel material is hermetically sealed within highly corrosion resistant zirconium alloy (or stainless steel) cladding tubes with welded end closures. The only mechanism available under normal operating conditions for radionuclides in spent fuel to become available for dispersal is through the corrosion of defective fuel pins by the pool waters. Experience at pools where aged fuel has been stored (GE Morris Operation and NFS West Valley) has shown that the activity level of the pool water does show an increase when more fuel is added to a pool but that the activity decreases rapidly with time. The apparent explanation is that only the fuel directly exposed by a cladding defect is available for attack and only for a relatively short time.

A Zircaloy-clad fuel bundle containing two failed rods was placed in a closed can after burnup of 1900 MWD/MTU. After nine years, the radioactive content of the water inside the can had risen to only 1 mCi ($\sim$ 5 ppm of ^{137}Cs).[23]

NFS reported[24] an experienced pool water impurities composition of 76% ^{137}Cs; 6% ^{134}Cs; 6% ^{124}Sb; 6% ^{144}Ce and 1% ^{90}Sr. GE Morris Operation has also identified ^{60}Co as a minor contaminant in pool waters. Because of the direct relationship between pool water activity levels and occupational exposures, there is an incentive to keep pool water activity levels under control at all times; values in the range of 10^{-4} to 10^{-3} μCi/ml are common.

4.2.2.5 Surface Crud Deposits

Crud deposits have been observed on the surfaces of fuel pins and fuel assembly hardware, particularly on the inner lower nozzle surfaces. The thickness of these crud layers varies from almost nil up to about 150 microns.[25] Surface appearance varies from a dense black for PWR fuels to an orange-red for some BWR fuels, dependng upon reactor primary coolant circuit characteristics. These crud layers are oxides of iron, nickel, and copper and mixed oxides.

These crud deposits slough off during shipping and are the principal source of contamination of cask coolants. A small fraction also apparently becomes either dissolved or suspended in the pool waters, e.g., ^{60}Co. However, based on visual observations at the NFS West Valley plant, most of the crud deposits remained on the fuel assembly until it was chopped up prior to reprocessing.

4.2.2.6 Airborne Radioactivity

Airborne radioactivity within a spent fuel storage facility is a function of: the pool water activity, care used in handling fuel, frequency of fuel transfer operations and good housekeeping practices. Based on G.E. experience, the airborne activity levels are a factor of 10^{-8} less than the pool water activity and are routinely less than 1% of the occupational exposure limits in 10 CFR Part 20, Appendix B, Table I.

4.2.2.7 Pool Water Purification System

Spent fuel storage pools are serviced by a pool water cleanup system consisting of filters and ion exchange units, and the necessary pumps, tanks and piping. These systems may contain concentrations of radionuclides as much as 100 times that of the pool waters, enough to require local shielding and carefully controlled operating procedures. However, the inventory of radionuclides available for disposal is limited to that contained in a spent filter or ion exchange unit at the time of replacement. As these are wetted materials, spills could cause a local decontamination and cleanup problem but the materials involved are readily contained.

4.2.2.8 Decontamination Solutions

Shipping casks represent the major source of contaminated wash solutions. During shipment some of the surface crud on fuel assemblies can become dislodged and become a source of contamination to the cask cavity. On receipt at the storage installation, the water in the cask cavity is sampled for radioactivity and, if necessary, flushed out before the cask is opened. The wash waters generated are collected in the onsite low level waste system for treatment prior to disposal.

Wash solutions from plant decontamination operations are also collected in the low level waste system for treatment prior to disposal.

The GE Morris Operation has a somewhat unique system, different from that described above. This facility has a vault which is embedded in rock for their collection of low level wastes. This vault was originally intended for the collection of low level wastes from the reprocessing plant and is designed for relatively long period onsite storage to take advantage of radioactive decay before final treatment and disposal. It is not anticipated that a storage only facility would be equipped with such a vault, but would more likely use relatively small volume tankage behind shielding for the collection of low level wastes prior to treatment.

4.2.2.9 Dry Waste Materials

A spent fuel storage operation also generates dry radioactive waste materials. These consist of contaminated protective clothing, blotting paper, and cleaning mops and plastic sheeting. Such materials are normally collected in plastic bags and packaged in drums prior to disposal. The contained radioactivity in such drums is normally in the order of 200 μCi/drum. This activity adheres to the materials involved and is not in a readily dispersable form.

4.2.2.10 Release Mechanisms

As underwater storage is a low temperature, low pressure environment, there is no driving force for the sudden release of a major fraction of the radioactive materials contained in the stored spent fuel even under abnormal operating conditions. Small quantities of radioactive materials could be released inside the facility during an inadvertent venting of a shipping cask while it is being prepared for unloading or a spill of low level waste materials in the waste handling and treatment system.

4.2.2.11 Offsite Dispersal Mechanisms

Again, because of the absence of high temperatures or pressures in an under water spent fuel storage operation, the only mechanism for offsite dispersal of released radioactive materials is atmospheric conditions.

4.2.3 <u>Accidents and Natural Phenomena</u>

For an accident to represent a potential radiological hazard to the general public, the same conditions apply - radioactive materials must be released from the facility and dispersed offsite. For this to happen:

- The radioactive materials involved must be rendered into a dispersable form,
- These must be released from the facility, and
- The conditions must be present for dispersion offsite of such released materials.

A range of potential accidents and natural phenomena events have been analyzed.

4.2.3.1 Accidents Resulting in Rupturing of Fuel Pins

Both NFS and AGNS included in their safety analysis reports (Docket Nos. 50-201 and 70-1729 respectively) an under water fuel drop accident in which it was assumed that all of the fuel pins in a fuel assembly were ruptured. Because of the age of the spent fuel, very little ^{131}I remains and with a decontamination factor of 100 for an under water release, a negligible amount of ^{131}I would be available for dispersion offsite. The NFS calculated release rates for an assembly exposed for 33,000 MWD/MTU and cooled for a minimum of 120 days were:

	Release Rate -- Ci/Sec	
Nuclide	From Fuel	From Pool
^{85}Kr	5.5×10^{-4}	5.5×10^{-4}
^{131}Xe	9.2×10^{-7}	9.2×10^{-7}
^{129}I	3.7×10^{-10}	3.7×10^{-12}
^{131}I	2.9×10^{-7}	2.9×10^{-9}

With ground level release dispersion factors in the order of 10^{-4} to 10^{-7} sec/m^3 at most sites, site boundary concentrations would be a small fraction of the 10 CFR Part 20, Appendix B, Column II, limits.

4.2.3.2 Low-Probability Missile Accident

An analysis has also been made of a low-probability missile accident at a storage only type of facility containing 1 year and 3 year, aged, spent fuel. The accident was defined as the penetration of the building by a tornado generated missile that lands in the storage pool. The activity in the gap between the fuel and the fuel cladding is released from the fuel pins ruptured by the impact of the missile. The missile evaluated was a 13.5-inch-diameter by 35-foot-long utility pole, travelling at 144 mph.

Assuming that the missile entered the pool at an optimum angle, a 45 foot row of fuel assemblies could be impacted if the missile was not deflected from its course of travel. Assuming a uniform storage array of 40 BWR assemblies and 27 PWR assemblies, a total of 20 MT of fuel could be impacted. It was assumed that 10% (a high figure) of the contained ^{85}Kr is in the fuel cladding gap and hence available for release. Similarly, 1% of the ^{129}I is also assumed present in the gap. However, iodine is soluble in water and an underwater release would be subject to a decontamination factor of at least 100. On this basis the source terms for spent fuel exposed to an average of 28,000 MWd/MTU shown in Table 4.5 were calculated.

Assuming an atmospheric dispersion factor (χ/Q) of 10^{-4} sec/m^3 for a ground level release and a site boundary distance of 275 meters, the calculated dose rates are shown in Table 4.6.

The calculated doses shown in Table 4.6 are obviously quite small and are a fraction of the average annual natural background dose of greater than 0.1 rem.

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Table 4.5. Calculated Source Terms for Low-Probability Missile Accident Analysis - Away-from-Reactor Storage Pool

Radio-nuclide	Inventory Ci/MT* 1 yr decay	Inventory Ci/MT* 3 yr decay	Fraction in Gap**	Release Fractions***	Curies Released per MTU 1-yr old fuel	Curies Released per MTU 3-yr old fuel	Curies Released per 20 MT of Fuel 1 yr old	Curies Released per 20 MT of Fuel 3 yr old
^{85}Kr	9.6×10^{3}	8.4×10^{3}	0.1	0.1	9.6×10^{2}	8.4×10^{2}	1.9×10^{4}	1.7×10^{4}
^{129}I	3.1×10^{-2}	3.1×10^{-2}	0.01	0.01	3.1×10^{-6}	3.1×10^{-6}	6.2×10^{-5}	6.2×10^{-5}

Bases:

*28,000 (average) MWd/MTU burnup, ORIGEN Code calculation.

**^{85}Kr = 10%; ^{129}I = 1%

***^{85}Kr = 100%; ^{129}I = 1% of gap activity

Table 4.6. Calculated Site Boundary Dose Rates for Low-Probability Missile Accident at Away-From-Reactor Storage Pool

Radio-nuclide	Ci Released 1 yr decay	Ci Released 3 yr decay	Exposure at Site Boundary, Ci-sec/m^3 1 yr decay	Exposure at Site Boundary, Ci-sec/m^3 3 yr decay	Dose Conversion Factor, Rem/Ci-sec/m^3	Critical Organ Dose, rem 1 yr decay	Critical Organ Dose, rem 3 yr decay
^{85}Kr	1.9×10^{4}	1.7×10^{4}	1.9	1.7	3.0×10^{-2}	5.7×10^{-2}**	5.1×10^{-2}**
^{129}I	6.2×10^{-5}	6.2×10^{-5}	6×10^{-9}	6×10^{-9}	4.6×10^{6}	2.9×10^{-2}***	2.9×10^{-2}***

*50-year commitment

**Skin

***Thyroid

4.2.3.3 Fires and Explosions

Fires and explosions could be the driving force for the dispersion of radioactive materials in finely divided forms. However, there is no need for the use of explosive materials in an AFR storage facility and normal operating procedures limit the accumulation of combustible materials such as paper. Such materials are used for routine decontamination operations, but as soon as used, these materials must be properly bagged to prevent a further spread of contamination. Serious fires and explosions are not considered credible in an AFR storage facility.

4.2.3.4 Criticality Accident

Assuming the fuel storage design was adequate, a criticality accident in a spent fuel pool could conceivably approach the power levels (less than 1,000 kW) of a "swimming pool" type of research reactor.[26] As proven by the operation of such reactors for many years, conditions did not generate enough energy to disperse any radioactive materials to the atmosphere from under more than 12 feet of water.

4.2.3.5 High Pool Water Activity

Based on operating experience at the GE Morris Operation and the NFS West Valley Plant, spent fuel storage pool water activity should normally be maintained at less than 5×10^{-3} µCi/ml. At this concentration the dose rate on the bridge crane above the pool is less than 2 mrem/hr.

An increase in the pool water activity by a factor of ~10 times to about 5×10^{-2} µCi/ml would result in a dose rate of about 20 mrem/hr based on NFS experience when their pool became contaminated due to ruptured metal fuel elements from the dual purpose N-reactor at Hanford.

During a period of high pool water activity, fuel transfer activities would normally be curtailed until the pool water activity is reduced to normal operating levels.

4.2.3.6 Rupture of Waste Tank or Piping

One of the potential sources of in-plant personnel exposure is the low level waste treatment system. The backwashes from the pool water filters and demineralizers are normally piped to a collection tank prior to concentration and solidification. Activity levels in the piping and collection tanks are in the order of 0.5 to 1.0 µCi/ml. For this reason, this system is normally located behind shielding.

A break in the piping or a rupture of the collection tank might cause a leak of 100 gals. of contaminated water to the floor inside the building. The area would have to be isolated, and decontamination and cleanup action initiated.

One method of cleanup would be to absorb the spillage with vermiculite and load it into drums for disposal. If the waste treatment facility is located within a shielded cell with a HEPA filter in its exhaust air duct, and only particulates are involved, 99.9% of which would be captured on the HEPA filter, the effects of the spill would be confined to the cell. A decontamination and cleanup operation would be necessary, but this could be confined and would have a negligible effect on the rest of the installation or its environs.

If the waste treatment facility is located behind shielding but not in an enclosed cell, or the cell door was open, the airborne fraction of the spill could be distributed within the facility in a pattern depending on air flow.

With an air volume of 100,000 ft^3 or greater, the activity of the building air might be increased initially, but with circulation through a HEPA filter, this activity could be reduced to normal levels within a short time. Access to the building could be restricted for this short period of time but essential operations could be carried out under "special work permit" restrictions.

Exposure of in-plant personnel should be readily controllable by operating procedures and physical barriers. There should be a negligible effect offsite.

4.2.3.7 Lowering of Pool Water Level

A 1,000-ton-capacity storage pool is estimated to contain 1,000,000 gallons of water and be 30 or more feet deep. The water in a spent fuel storage pool serves the dual functions of heat removal and shielding. Spent fuel storage pools are normally designed with a minimum of 12 feet of water over the fuel in storage, enough to reduce the gamma dose rate from the fuel assemblies to less than 0.5 mr/hr at the pool surface.

Fuel transfer mechanisms have limit switches and mechanical stops to prevent raising a fuel element or a storage canister to less than 9 or 10 feet of the water surface.

- A loss of 5% of the water, about 50,000 gallons, would have only a negligible impact on personnel exposures,
- A loss of 25% of the water, about 250,000 gallons, would reduce the shielding over the stored fuel to about 6 feet. Under these conditions the fuel transfer bridge crane work could be carried on within the facility but this may have to be done under "special work permit" conditions.

The fall of the water level to this depth may require an emergency modification of the cooling water circuit inlet and outlet lines, such as connecting emergency supply and cutting off any bleed-off system, but this should be feasible without serious over exposure of personnel.

While the loss of all water is beyond the design basis envelope, it involves only low risks for independent spent fuel storage installations in which only aged spent fuel is stored. The major consequence of such an unlikely event would be a small skyshine dose at a site boundary. Dose rate versus distance calculations have been made for this event.[27]

The heat generation rate of spent fuel decreases rapidly with time for a short period following discharge from a reactor. For example, at one year after discharge the spent fuel heat generation rate is less than one percent of its rate when it is discharged from the reactor. At ten years its heat generation rate has decreased by another factor of ten to one-tenth of one percent.

Assuming that the spent fuel stored at an independent spent fuel storage installation is at least one year old, calculations have been performed to show that loss of water should not result in fuel failure due to high temperatures if proper rack design is employed.[28] Such design specification is included in NRC regulatory guidance now in preparation. Cooling by natural convection air currents alone should be adequate. The staff believes that such storage facilities can be designed and constructed to assure that loss of the pool water will be a highly unlikely event. Based on its safety reviews of similar facilities the staff finds that such pools can be constructed to withstand severe events and backup sources of water can be provided.

4.2.3.8 Loss of Cooling

Because there is adequate time to take corrective action in the event of a loss of cooling at an AFR storage facility, there are no special requirements placed on the design and construction of the cooling system other than the pool water be circulated in a closed loop. However, in the course of a safety review, the staff does require an adequate backup supply of water. A loss of the cooling system for a number of weeks was experienced at the GE Morris facility operating during the 1976-1977 winter with no adverse effects.

On January 16, 1977 a two hour interruption in the power supply shut down the circulating pump. The outdoor temperature was -19°F. When normal flow was reestablished, a pipe break was discovered and the system was shut down and drained. With 225 tons of fuel in storage, the GE pool

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reached an equilibrium temperature of 115°F over a number of weeks. The humidity in the building was uncomfortably high, but otherwise this incident had no adverse impact on either plant personnel or the general public.

NFS showed an analysis in their SAR for a planned expansion program of their pool filled with fuel (giving off 12×10^6 Btu/hr) and allowed to reach a boiling temperature. Their calculated time required to reach boiling was 48 hours for an isolated pool, and a boil off rate of 1,500 gal/hr. A comparable staff calculation for a much larger pool and more compact fuel storage but with a heat generation rate more typical of fuel placed in extended storage showed a temperature rise of about 4°F/hr. and the time to reach boiling was 33 hours.

These figures show that there is time to take corrective action even with a complete loss of cooling. If conditions preclude reactivation of the cooling system within the time allowance to reach boiling, makeup water must be provided to offset evaporation losses. A staff calculation for a pool containing 1,000 tons of fuel with a heat generation rate of 3.4×10^7 Btu/hr would require 60 gal/min to maintain the water level under boiling conditions.

To assure the availability of makeup water during an extended outage of the cooling system, there must be a reliable water source and a means of delivering water to the spent fuel storage pools should the need arise.

NFS calculated that, with a decontamination factor of 10^4, the airborne activity within the building, with the pool water boiling, would be less than the occupational exposure concentration limits shown in 10 CFR Part 20, Appendix B, Table II, Column I.

4.2.4 <u>Considerations and Assumptions Used for Offsite Transportation Accident Analysis</u>

All information in this section is summarized from WASH-1238, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants."[29] The consequences of a major release of radioactive material from a spent fuel shipping cask could be severe; however, the low probability of such an occurrence during transportation makes the risk from such accidents extremely small. Spent fuel shipping casks are designed to withstand severe transportation accidents without significant loss of contents or increase in external radiation levels. The casks are protected from the damaging effects of impact, puncture, and fire by thick outer plates, protective crash frames, or other protective features designed to control damage.

Transportation accidents occur in a range of frequencies and severities. Most accidents occur at low vehicle speeds. The severity of accidents is greater at higher speeds, but the frequency decreases as the severity increases. Transportation accidents usually involve some combination of impact, puncture, fire, or submersion in water.

4.2.4.1 Estimates of Releases in Accidents

Estimates of the amount of radioactive material released and the calculated doses in the unlikely event that a shipping cask is breached are summarized herein. The consequences in terms of potential doses to humans were calculated for the estimated releases of ^{85}Kr, ^{131}I, and fission products. Normal distributions of weather and population densities for a release on land were used in the calculations.

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PLANNING BASIS FOR THE DEVELOPMENT OF STATE AND LOCAL GOVERNMENT RADIOLOGICAL EMERGENCY RESPONSE PLANS IN SUPPORT OF LIGHT WATER NUCLEAR POWER PLANTS

A Report Prepared by a
U. S. Nuclear Regulatory Commission and
U. S. Environmental Protection Agency
Task Force on Emergency Planning

DECEMBER 1978

[Appendix I, pages
I-1 - I-11 Follows]

Office of State Programs
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Office of Radiation Programs
U. S. Environmental Protection Agency

APPENDIX I

RATIONALE FOR THE PLANNING BASIS

A. General Considerations

The Task Force considered various rationales for establishing a planning basis; including risk, probability, cost effectiveness, and consequence spectrum. After studying the various approaches discussed below, the Task Force chose to base the rationale for the planning basis on a spectrum of consequences, tempered by probability considerations.

With respect to the risk* rationale, such an approach would establish "planning guidance" that could be compa with the risks associated with non-nuclear accidents. This rationale would seemingly give a uniform basis for emergency anning and would clearly indicate the level of risk that could be mitigated by advanced planning. However, emergency planning for non-nuclear hazards is not based upon quantified risk analyses. Risk is not generally thought of in terms of probabilities and consequences, rather it is an intuitive feeling of the threat posed to the public. Reactors are unique in this regard: radiation tends to be perceived as more dangerous than other hazards because the nature of radiation effects are less commonly

*Risk is defined as accident consequences times the probability of accident occurrence.

understood and the public generally associates radiation effects with the fear of nuclear weapons effects. In addition, a risk-related rationale might imply the determination of an acceptable level of risk which is outside the scope of the Task Force effort. Choosing a risk comparable to non-nuclear events, therefore, was not directly used as the rationale for an emergency planning basis.

With respect to a probability rationale, one could arrive at "planning guidance" by selecting an accident probability below which development of an emergency plan could not be justified. Factors favoring using this rationale center around providing a quantitative probability basis, which could be compared with the probabilities of other types of emergencies for which plans are prepared.

Factors arguing against the probability rationale are similar to those against the risk approach. Emergency planning is not based upon quantified probabilities of incidents or accidents. On the basis of the accident probabilities presented in the Reactor Safety Study (nuclear and non-nuclear) society tolerates much more probable non-nuclear events with similar consequence spectrums without any specific planning. Radiological emergency planning is not based upon probabilities, but on public perceptions of the problem and what could be done to protect health and safety. In essence, it is a matter of prudence rather than necessity.

A generic "probability of an event" appropriate for planning has many implications felt to be outside the scope of the Task Force objective. However, the concept of accident probability is important and does have a place in terms of evaluating the range of the consequences of accident sequences and setting some reasonable bounds on the planning basis. The probability rationale was used by the Task Force to gain additional perspective on the planning basis finally chosen.

With respect to a cost-effectiveness rationale, the level of emergency planning effort would be based on an analysis of what it costs to develop different levels of such a plan and the potential consequences that could be averted by that degree of development. The factor favoring the cost-effectiveness rationale is that an emergency plan could be developed on the basis of cost per potential health effect averted. Factors arguing against the cost-effectiveness rationale are the difficulty in arriving at costs of plan development and maintenance and considerations that general and radiological emergency response plans have already been developed. In addition, absent an actual accident, it would be very difficult to assign a dollar value to the effectiveness of the plan in terms of health effects averted.

Lastly, the calculated consequences from a spectrum of postulated accidents was considered as the rationale for the planning basis.

Such a rationale could be used to help identify desirable planning elements and establish bounds on the planning effort. Further, a planning basis could be easily stated and understood in terms of the areas or distances, time frames and radiological characteristics that would correspond to the consequences from a range of possible accidents. Consequence oriented guidance would also provide a consistency and uniformity in the amount of planning recommended to State and local governments. The Task Force therefore judged that the consequences of a spectrum of accidents should be the principal rationale behind the planning basis.

B. Consequence Considerations

The Task Force considered the complete spectrum of accidents postulated for various purposes, including those discussed in environmental reports (i.e. best estimate Class 1 through 8 accidents), accidents postulated for purposes of evaluating plant designs (e.g. the DBA/LOCA), and the spectrum of accidents assessed by the Reactor Safety Study. The Task Force concluded that the environmental report discussions (Class 1-8) were too limited in scope and detail to be useful in emergency planning.

1. Design Basis Accidents

Under NRC Regulations, the site/reactor design combination must be such that the consequences of design basis accidents are below the plume exposure guidelines of 10 CFR Part 100. The design basis loss-of-coolant accident (DBA-LOCA) has been typically the most severe design basis accident in that it results in the largest calculated offsite doses of any accident in this class. The DBA-LOCA is not a realistic accident scenario in that the release magnitudes are much more severe than would be realistically expected and may exceed that of some core-melt type accidents. A best estimate assessment of the release following a LOCA would be significantly smaller than the DBA-LOCA used for siting purposes. An analysis of this accident has been performed for most of the power plants licensed or under review by NRC to determine the dose/distance relationships as computed by traditionally conservative assumptions used under 10 CFR Part 100 requirements. Results of this study are presented later in this appendix. The study concluded that the higher PAG plume exposures of 25 rem (thyroid) and 5 rem (whole body) would not be exceeded beyond 10 miles for any site analyzed. Even under the most restrictive PAG plume exposure values of 5 rem to the thyroid and 1 rem whole body, over 70 percent of the plants would not require any consideration of emergency responses beyond 10 miles. It should be noted that even for the DBA-LOCA, the lower range of the plume PAGs would likely not be exceeded outside the low population zone (LPZ) for average meteorological conditions.

For the ingestion pathways, under the same DBA-LOCA conditions, the downwind range within which a PAG of 1.5 rem thyroid could be exceeded would be limited to within 50 miles even under the conservative 10 CFR 100 assumptions. The 50 mile distance is also justified as a maximum planning distance because of likely significant wind shifts within this distance that would further restrict the radius of the spread of radioactive material.

2. Class 9 Accidents

"Class 9" accidents cover a full spectrum of releases which range from those accidents which are of the same order as the DBA-LOCA type of releases; i.e., doses on the order of PAGs within 10 miles; to those accidents which release significant fractions of the available radioactive materials in the reactor to the atmosphere, thus having potential for life-threatening doses. The lower range of the spectrum would include accidents in which a core "melt-through" of the containment would occur. As in the DBA-LOCA class, the doses from "melt-through" releases (involving thousands of curies) generally would not exceed even the most restrictive PAG beyond about 10 miles from a power plant. The upper range of the core-melt accidents is categorized by those in which the containment catastrophically fails and releases large quantities of radioactive materials directly to the atmosphere because of over-pressurization or a steam explosion. These accidents have the potential to release very large quantities (hundreds of millions of curies) of radioactive materials. There is a full spectrum of releases between the lower and upper range with all of these releases involving some combination of atmospheric and melt-through accidents. These very severe accidents have the potential for causing serious injuries and deaths. Therefore, emergency response for these conditions must have as its first priority the reduction of early severe health effects. Studies[6,7] have been performed which indicate that if emergency actions such as sheltering or evacuation were taken within about 10 miles of a power plant, there would be significant savings of early injuries and deaths from even the most "severe" atmospheric releases.

For the ingestion pathways, (due to the airborne releases and under Class 9 accident conditions), the downwind range within which significant contamination could occur would generally be limited to about 50 miles from a power plant, because of wind shifts during the release and travel periods. There may also be conversion of iodine in the atmosphere (for long time periods) to chemical forms which do not readily enter the ingestion pathway. Additionally, much of the particulate materials in a cloud would have been deposited on the ground within about 50 miles.

C. Probability Considerations

An additional perspective can be gained when the planning basis is considered in terms of the likelihood (probability) of accidents which could require some emergency response.

Probabilities can be used to give a perspective to the emergency planner by comparing the chance of a reactor accident to other emergencies for which plans and action may be required. This consideration forms an additional basis upon which the Task Force selected the planning basis. The Reactor Safety Study (RSS) estimated the probabilities* of various severe accidents occurring at nuclear power plants. The probability of a loss-of-coolant accident (LOCA) from a large pipe break was estimated to be approximately one chance in 10,000 (1×10^{-4}) of occurring per reactor-year. LOCA accidents would not necessarily lead to the melting of the reactor core since emergency core cooling systems (ECCS) are designed to protect the core in such an event. In fact, other accident initiating events such as the loss-of-coolant accident from a small pipe break or transient events have a higher chance of leading to core-melting than do large LOCA accidents. Core-melt type accidents were calculated to have a probability of about one chance in 20,000 of occurring per reactor-year. There is a significant degree of uncertainty associated with both of the above probability estimates.

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* Use of the RSS probability estimates, in the context of emergency planning, has been thoroughly examined. It is recognized that there is a large range of uncertainties in these numbers (as indicated in the Risk Assessment Review Group Report, NUREG/CR-0400), but the perspective gained when considering the probabilities is important in making a rational decision concerning a basis for emergency planning.

The degree of uncertainty is such that no differentiation can be confidently made, on a probabilistic basis, between the DBA/LOCA and the releases associated with less severe core-melt categories.

As discussed in Appendix III, the Task Force has concluded that both the design basis accidents and less severe core-melt accidents should be considered when selecting a basis for planning pre-determined protective actions and that certain features of the more severe core-melt accidents should be considered in planning to assure that some capability exists to reduce the consequences of even the most severe accidents. The low probabilities associated with core-melt reactor accidents (e.g. one chance in 20,000 or 5×10^{-5} per reactor-year) are not easy to comprehend and additional perspectives are useful. Within the next few years, there will have been accumulated approximately 500 reactor-years of civilian nuclear power plant operation in this country. Less than 30% of all core melt accidents would result in high exposure outside the recommended planning distances. Therefore, over this time period* the probability of an accident within the USA with exposures exceeding the plume or ingestion PAGs <u>outside</u> the planning basis distances would be about 1.5×10^{-5}** x 500 or about 1 chance in

* The Reactor Safety Study explicitly limits its analyses to the first 100 reactors and five years (through 1980).

** This estimate is based upon the assumptions of the RSS. It should be noted that there is a large uncertainty on this number.

100. To restate this, there is about a 1% chance of emergency plans being activated in the U.S. beyond the recommended EPZs within the next few years. For a single State, this probability drops appreciably. For a State with ten reactors within or adjacent to its borders, the probability of exceeding PAGs outside the planning basis radius for the plume exposure pathway is about $1.5 \times 10^{-5} \times 10$ or about one chance in 6000 per year according to the Reactor Safety Study analysis.

For perspective, a comparison between reactor accidents and other emergency situations can be made. Considerations of emergency planning for reactor accidents are quite similar to many other emergencies; floods, for example, have many characteristics which are comparable. Timing, response measures and potential consequences, such as property damage are similar for both events.

Flood risk analysis has been carried out by the Flood Insurance Program of the Department of Housing and Urban Development and the Corps of Engineers. Flood plains have been designated for all areas of the country by computing the probability of being flooded within a certain period of time; ie., the 100-year flood plain designates those areas which can be expected to be under water when the worst flood in a century occurs. Even with this relatively high probability of severe flood occurrence there are no explicit requirements for emergency response planning.

Hurricanes and tornadoes are two potential threats for which some emergency planning is required. Approximately 2 hurricanes per year may be expected to hit the Atlantic coastal States which require emergency response. For individual States, the hurricane frequency ranges from 0.01 to 0.65 per year. Tornadoes have a very high probability of occurrence per year. A severe tornado can be characterized by wind speeds of over 200 miles per hour. Such tornadoes are capable of lifting cars off the ground, tearing roofs and walls off frame houses, overturning trains, and uprooting or snapping most trees. Emergency actions would probably be taken for such tornadoes. The frequency of severe tornadoes for individual States, ranges from about 0.1 to 4 per year.

Severe reactor accidents are at least 100 times less likely to occur than these other disasters requiring emergency response. We nevertheless believe, that it is appropriate to develop flexible emergency response capabilities which will assure that consequences from nuclear reactor accidents are minimized.