UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	
Vermont Yankee Nuclear Power Corporation	Docket No. 50-271
(Vermont Yankee Nuclear Power Station)	

NEW ENGLAND COALITION ON NUCLEAR POLLUTION'S
SUPPLEMENTAL RESPONSE TO VERMONT YANKEE
SPENT FUEL POOL EXPANSION REQUEST,
51 FED. REG. 22,245

On July 21, 1986, The New England Coalition on Nuclear Pollution ("NECNP") filed an initial response to the Vermont Yankee request to expand spent fuel storage capacity, noticed at 51 Fed. Reg. 22,245 (June 18, 1986). NECNP also requested an extension to supplement that response. The following constitutes NECNP's Supplemental Response.

When originally licensed, the Vermont Yankee spent fuel storage pool capacity was 600 spent fuel assemblies. Yankee maintained that this capacity was adequate since fuel would be stored for only a year or two onsite and then shipped away for processing. In 1977, Yankee received a license amendment authorizing an increase in spent fuel storage capacity to 2000 assemblies, which it states is adequate, with full core discharge reserve space, until 1990. Yankee now seeks another increase to 2,870 fuel assemblies, to be accomplished by removing the storage racks and replacing them with racks spaced more tightly together, greatly increasing the density of the stored assemblies. To our knowledge, Yankee has never removed any spent fuel from the site, and with this latest request, seeks to complete a near quintupling of the originally licensed authority to store spent fuel.

NRC, in turn proposes to approve this latest license amendment request without opportunity for prior hearing on the asserted grounds that it presents "no significant hazards consideration." 51 Fed. Reg. 22,245, 22,246, Col. 1, June 18, 1986. Moreover, insofar as we are able to determine, NRC has made no review of the environmental impact of the proposed action, nor considered whether alternatives - including, for example, dry case storage - present significant safety and environmental advantages over increasing the density of the pool. NECNP contends that this license amendment does present a "significant hazards consideration;" that is, that it raises a substantial safety and environmental question.

In addition, the proposal requires an Environmental Impact Statement pursuant to the National Environmental Policy Act ("NEPA").

STORAGE EXPANSION AND DENSIFICATION SIGNIFICANTLY INCREASES THE RISK OF ACCIDENT

In order to prevent the more tightly packed fuel assemblies from beginning a nuclear reaction (i.e. to keep the assemblies "subcritical"), it is necessary for the company to surround each with a neutron absorbing material.

At 2,870 assemblies, the Vermont Yankee pool would be capable of storing almost <u>eight</u> full core loads. (A full core at the plant is 368 assemblies). This obviously constitutes a very large quantity of long-lived radioactivity which, if released, could lead to substantial environmental contamination.

The driving force for such a release can be created because the neutron-absorbing material, necessary to prevent criticality in the pool, would also act to suppress heat transfer from the spent fuel in the event of water loss from the pool. This can lead cladding temperatures to rise high enough to initiate zirconium-air or zirconium-steam reactions, creating heat sufficient to provide a driving force which can release volatile radionuclides from the fuel. See, The Source Term Debate, A Report by the Union of Concerned Scientists, Sec. 9.5.2, p. 9-24, Jan., 1986. These would include, for example Cesium-137, a very long-lived element. The surrounding reactor building is not designed to withstand an explosion of hydrogen generated in the zirconium-steam reactor. Thus, a significant release to the environment might occur.

The Union of Concerned Scientists analysis further states:

> A severe reactor accident could lead to loss of water from the spent fuel pool in two ways. First, violent phenomena such as hydrogen explosions could lead to a breach of the pool. This would be most significant for those plants (such as Mark I and II BWRs) where the pool is above grade level. Second, the pool cooling systems may be disabled as a part of the reactor accident sequence. Repair of these systems might then be precluded for several weeks or longer, due to high radiation fields around the plant. Water would then be lost by evaporation, leading to uncovering of the spent fuel in times of the order of a week or two (the time depending heavily on the age after discharge of the most recently discharge spent fuel).

Id. Sandia National Laboratories performed calculations to estimate the temperatures which could be recorded in a typical pool in the event of loss of the water. While Sandia did not analyze the worst case, its calculation still showed that cladding temperatures could exceed 1000°C. At this temperature, both the zirconium-air and zirconium-steam reactions proceed vigorously. A.S. Benjamin, et al. Spent Fuel Heatup Following Loss of Water During Storage, Sandia National Laboratories, NUREG/CR-0619, Mar., 1969. In addition, a recent NRC-sponsored experimental and theoretical research study concluded that, if the most recently discharged fuel begins a self-sustaining zirconium oxidation, the heat so generated can raise the temperature of surrounding assemblies to the point of ignition. In this way, the "fire" may travel throughout the entire fuel pool. N.A. Pisano, et al., The Potential for Propogation of a Self-Sustaining Zirconium Oxidation Following Loss of Water In a Spent Fuel Storage Pool, Sandia National Laboratories, Draft Report, Jan., 1984.

Over the past eight years a body of evidence and scientific opinion has been growing, such as that summarized in the Union of Concerned Scientists report, which raise serious questions about the safety of spent fuel pool storage. These concerns are greatest in the case of high-density racking and particularly in plant designs, such as Vermont Yankee, where the spent fuel pool is located above ground level. While a spent fuel pool release would not happen quickly after the loss of water, it is not correct to assume from this that water could necessarily be restored to the pool in all cases. The most probable circumstances for release of the radioactive contained in the pool are those associated with a severe reactor accident. Such an accident could involve fire or explosion in or outside the containment and/or release of radiation. Even were the release not at the worst end of the possible spectrum as far as contamination of the outside environment is concerned, it could be severe enough to prevent access to the spent fuel pool. The assemblies would continue to heat up and enter the exothermic reactions described above.

Copies are attached of testimony presented on these issues by Dr. Gordon Tompson before the Sizewell "B" Public Inquiry in England in February, 1984 and before the Minnesota Energy Agency in 1979 concerning the Prairie Island spent fuel pool expansion. While both plants in question are pressurized water reactors (and thus the scenario resulting in a loss of water from the pool would differ) the discussion of the physical phenomena involved in a release of readioactivity from the spent fuel pool is applicable. Indeed, as the Union of Concerned Scientists report quoted above notes, the risk would appear to be greater for reactors of the Vermont Yankee design than for pressurized water reactors because the pool is above grade. Moreover, the evidence is strong that the likelihood of a large release of radioactivity in the event of severe accident, blocking access to the pool, is greatest for GE Mark I plants such as Vermont Yankee. NRC's current operative assumption, presented in a September 11, 1986 meeting between top NRC officials and the BWR owners group, is that 1 in 2 severe accidents in a Mark 1 plant will result in large releases.

In 1979, the Lower Saxony State Government set up an international review group to review and advise it regarding the application pending before it to build a nuclear storage, reprocessing, waste disposal and fuel fabrication facility at Gorleben, Germany. The resulting report was subject to public examination over a week of proceedings. After hearing the evidence, the Governor of the State disapproved the application and announced that modifications would be required as a condition of future re-application. The declaration of the Lower Saxony Government is attached. While concluding that the portion of the facility involving waste disposal in a salt dome did not pose unreasonable risk, the state government was unwilling to approve the spent fuel pool storage portion. The relevant chapter of the Report of the Goreben International Review is also enclosed, which analyzes the consequences of loss of cooling water to the spent fuel storage ponds.

In summary, the storage of a very large amount of radioactive material in the Vermont Yankee spent fuel pool constitutes a significant risk. That risk is obviously increased by expansion of the pool. Should a release occur, the magnitude of the consequences, particularly the greater contamination of land by long-lived Cesium-134 and the concomitant increase in latent health effects, could be much greater. The documents available in this docket contain no consideration whatever by NRC of these issues.

Moreover, there has been nothing approaching a rational consideration of the available alternatives, pursuant to NEPA. Yankee's "consideration" of the available technical alternatives, including dry cask storage, consists of the assertion that none has been licensed by another commercial utility. This hardly suffices under NEPA, particularly when use of such an alternative would greatly reduce both the probability and consequences of an accident.

THE PROPOSAL DOES PRESENT A SIGNIFICANT HAZARDS CONSIDERATION

NRC must offer an opportunity for a hearing prior to granting any licenses amendment except in cases involving "no significant hazards consideration." 42 U.S.C. §2239. This exception to the prior hearing requirement is contained in the 1982 "Sholly amendment" to the Atomic Energy Act. The legislative history of this amendment is replete with evidence that it was specifically intended by Congress that spent fuel re-racking such as this one would <u>not</u> be included within the "no significant hazards consideration" exception. The first reference to the subject occurred in the House of Representatives on November 5, 1981 when the House version of the bill (HR 4255) was considered and passed:

> Mrs. SNOWE. Would the gentleman anticipate this no significant hazards consideration would not apply to license amendments regarding the expansion of a nuclear reactor's spent fuel storage capacity of the reracking of spent fuel pools?
>
> Mr. OTTINGER. If the gentlewoman will yield, <u>the expansion of spent fuel pools and the reracking to the spent fuel pools are clearly matters which raise significant hazards considerations</u>, and thus amendments for such purposes could not, under Section 11 (a), be issued prior to the conduct or completion of any requested hearing or without advance notice. (127 Cong. Record H 8156) (emphasis added)

The Senate committee on Environment and Public Works repeated this belief in its report on S. 1207:

> The committee recognizes that reasonable persons may differ on whether a license amendment involves a significant hazards consideration. Therefore, the Committee expects the commission to develop and promulgate standards that, to the maximum extent practicable draw a clear distinction between license amendments that involve a significant hazards consideration and those that involve no significant hazards considerations. <u>The Committee anticipates, for example, that, consistent with prior practice, the Commission's</u>

> standards would not permit a "no significant hazards consideration" determination for license amendments to permit reracking to spent fuel pools.

Senate Report No. 97-113, U.S. Code Cong. & Ad. News p. 3599 (emphasis added).

Finally, Commissioner Asselstine (prior to his appointment) confirmed the existence of this practice in a response to Senator Mitchell:

> Senator Mitchell: There is, as you know, an application for a license amendment pending on a nuclear facility in Maine which deals with the reracking storage question. And am I correct in my understanding that the NRC has already found that such applications do present significant hazards considerations and therefore that petition and similar petitions would be unaffected by the proposed amendment?
>
> Mr. Asselstine: That is correct, Senator. The Commission has never been able to categorize the spent fuel storage as a no significant hazards consideration.

Transcript of meeting of Senate Committee on Env. & Public Works, quoted in March 15, 1983 letter from Senators Simpson, Hart, and Mitchell to Chairman Palladino.

It is therefore not unusual that the Conference Report on this legislation did not specifically mention reracking. The issue had been raised in each House and there had been complete agreement. Even the General Counsel and the Executive Legal Director, in a memorandum to Chairman Palladino and the Commissioners concluded

> In conclusion, we observe that although discussion of this issue is sparse, every reference on both the House and Senate sides reflects an understanding that expansion and reracking of spent fuel pools are matters which involve significant hazards considerations.

Moreover, the Conference report on the 1982 amendments emphasizes that if there is any doubt, the Commission should not make the "no significant hazards consideration" determination, but instead should permit a hearing before acting: "[The standards which the NRC promulgates to implement the amendments] should be capable of being applied with ease and certainty, and should ensure that the NRC staff does not resolve doubtful or borderline cases with a finding of no significant hazards consideration." House Conference Report No. 97-884, p. 37, reprinted in U.S. Code Cong. & Ad. News at 3607, (emphasis added). The Conference Report further emphasized its directive that the NRC was not to use the "no significant hazards consideration" determination in reviewing amendments involving irreversible consequences because such use would, as a practical matter, eliminate the public's right to a hearing:

> The conferees intend that in determining whether a proposed license amendment involves no significant hazards consideration, the Commission should be especially sensitive to the issue posed by license amendments that have irreversible consequences (such as those permitting an increase in the amount of effluents or radiation emitted from a facility or allowing a facility to operate for a period of time without full safety protections). In those cases, issuing the order in advance of a hearing would, as a practical matter, foreclose the public's right to have its views considered. In addition, the licensing board would often be unable to order any substantial review as a result of an after-the-fact hearing. Accordingly, the conferees intend the commission be sensitive to those license amendments which involved irreversible consequences.

Conference Report at p. 38, (emphasis added).

The legislative history demonstrates repeatedly that Congress sought to ensure full public participation before the amendment authorization when it enacted the 1982 amendments:

> The conference agreement maintains the requirement of the current section 189a. of the Atomic Energy Act that a hearing on the license amendment be held upon the request of any person whose interest may be affected. The agreement simply authorizes the Commission, in those cases where the amendment involved poses no significant hazards consideration, to

> issue the license amendment and allow it to take effect before this hearing is held or completed. The conferees intend that the Commission will use this authority carefully, applying it only to those license amendments which pose no significant hazards consideration.

Conference Report at p. 37, reprinted in U.S. Code Cong. & Ad. News at p. 3607 (emphasis added). Likewise, the Senate confirmed its intent that the public right to a hearing was not to be circumscribed with the new amendments:

> . . .the Committee expects the NRC to exercise its authority under this section only in the case of amendments not involving significant safety questions. Moreover, the Committee stresses its strong desire to preserve for the public a meaningful right to participate in decisions regarding the commercial use of nuclear power.

Senate Report at p. 14, reprinted in U.S. Code Cong. & Ad. News at p. 3598, emphasis added. And, as explained above, Congress explicitly directed that the Commission was to "ensure that the NRC staff does not resolve doubtful or borderline cases with a finding of no significant hazards consideration." This situation is not even a "borderline" case in light of the unusually explicit legislative history concerning spent fuel pool expansions.

Just last week, the United States Court of Appeals for the Ninth Circuit ruled that the NRC may not authorize spent fuel pool re-racking at the Diablo Canyon plant without offering a prior hearing. San Luis Obispo Mothers for Peace et al. v. NRC, No. 86-7297 (9th Cir. September 11, 1986). In interpreting the Sholly amendment, the court emphasized the "Congressional directive that doubts be resolved in favor of a prior hearing and that the NRC staff not prejudge the merits of a proposed licensed amendment." Id. at 8. Governed by this standard, the proposal raises significant hazards considerations.

Without conceding that NRC's rules properly implement the underlying law, they provide that a "no significant hazard consideration" finding is appropriate only if a proposed amendment does not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety

Keeping in mind the court's admonition that in applying this standard, NRC is <u>not</u> to prejudge the merits of the issues. NECNP contends that all three tests are met. As noted above, the expansion involves a significant increase in the consequences of reactor accidents. In particular, 9 10 CFR §50.44, concerning standards for combustible gas control, is predicated upon the assumption that core damage may occur. Thus, such an accident is "evaluated" for purposes of this rule. In the event of such an accident occurring at a time when Vermont Yankee is de-inerted, significant amounts of hydrogen would be generated.

Should such hydrogen be vented or otherwise released outside the containment into the building which houses the storage pool, and is not designed to withstand hydrogen expolosion, it could disable the spent fuel pool cooling systems or even threaten the structural integrity of the pool. Even a reactor accident which is not sufficiently severe to cause a significant release of fision products from the containment could involve the generation of explosive amounts of hydrogen, as in TMI-2. Indeed, during the TMI-2 accidents, a hydrogen explosion outside containment did occur.

Such a sequence of events both increases the consequences of an "evaluated" reactor accident and creates the possibility of a new or different kind of accident - a radioactive release from the spent fuel pool as described above and in the attachments. In addition, the storage of more fuel in the manner proposed, including the ability to emplace the freshest and hottest fuel, decreases a margin of safety by decreasing the time between the onset of heat-up of the fuel and release of radioactivity.

CONCLUSION

For the reasons stated above, NRC's determination that this proposal involves no significant hazards consideration is legally and factually insupportable. Under the Atomic Energy Act, Yankee's request requires NRC to provide an opportunity for a hearing before approval. NECNP would be interested in exploring with other interested parties the possibility of agreeing to informal procedures to govern such a proceeding.

In addition, this is a major federal action requiring compliance with the provisions of the National Environmental Policy Act. To this point, NRC has taken no steps to carry out its obligations under NEPA. These include, *inter alia*, the requirement to analyze and present for public comment the environmental consequences of a worst case accident, (40) CFR §1502.22, and to review the alternatives to this proposed action. NRC must do this before permitting the spent fuel storage expansion.

September 19, 1986

By [signature]

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LIST OF APPENDICES

APPENDIX A

SIZEWELL 'B' PUBLIC INQUIRY

Proof of Evidence on:

SAFETY AND WASTE MANAGEMENT IMPLICATIONS OF THE SIZEWELL PWR

On Behalf of the
Town and Country Planning Association

By: Gordon Thompson

With supporting evidence by: Steven Sholly

February 1984

Preface

This proof is a modified version of a report with the same title which has been submitted to the Inquiry as TCPA/S/127.

Modifications have been made to the overview of TCPA/S/127, but all the annexes are unchanged. Accordingly, this proof consists of an overview (modified from the overview of TCPA/S/127) plus annexes designated A through U (each of which is identical to the same annexe of TCPA/S/127).

Gordon Thompson, the principal author of TCPA/S/127, is the principal witness for this proof. He will be supported by Steven Sholly, who assisted in the preparation of TCPA/S/127. Five other consultants also contributed to TCPA/S/127, but none of those people is offered as a witness before the Inquiry.

Annex Q

RISKS ARISING FROM SPENT FUEL MANAGEMENT

prepared by Gordon Thompson
this version completed 30 November 1983

1. Introduction

In this context, "spent fuel management" refers to interim storage of spent fuel at the Sizewell site, or at another CEGB site, and its transportation. The risks associated with interim storage at a non-CEGB site, with reprocessing, and with final disposal, are not addressed here.

The CEGB proposes to store spent fuel, on an interim basis, in a water-filled pool adjacent to the containment building of the Sizewell PWR. Moreover, the Board is making provisions to eventually expand the pool's storage capacity, via high-density racking, to the equivalent of 7 reactor cores (21 years' discharge).

There is a risk associated with high-density racking. Loss of water from the pool can lead to overheating of the spent fuel and consequent release of radioactivity to the environment.

An alternative approach to interim storage, not subject to the same scenario, is dry storage. Considerable progress has been made in this area in recent years, in the UK and elsewhere.

During transport of spent fuel, there are also potential dangers. Through sabotage, accidental impact, or fire, it is possible for some of the radioactivity in the spent fuel to be released to the environment. The amount released would, of course, vary according to the severity of the accident.

This annex briefly addresses these issues. Section 2 discusses the risk associated with pool storage of spent fuel, while Section 3 discusses the alternative option of dry storage in casks. Finally, Section 4 addresses transport incidents.

2. Risks of Pool Storage

The CEGB plans an initial storage capacity of 324 fuel assemblies in the Sizewell PWR's spent fuel pool. Subsequently, this capacity can be expanded by installing high-density racks, to an ultimate capacity of 1377 fuel assemblies (7 reactor cores). In this high-density configuration, the centre-to-centre distance of the fuel assemblies will be about 10 inches. As the normal refuelling cycle involves discharge of 1/3 core annually, this 7-core capacity represents 21 years' discharge of spent fuel[Q.1].

In order to prevent criticality, which might arise at these high densities, each spent fuel assembly will be enclosed in a tube whose walls are made of neutron-absorbing material. Although effective at suppressing criticality, those tubes introduce a new hazard. In the event of water loss from the pool, the spent fuel can overheat.

Figure Q.1 shows some estimates, from a study performed at Sandia Laboratories, of clad temperature in the event of water loss from a pool containing spent fuel in a high-density configuration. The most serious case is the

"Blocked Inlets" case, wherein the convective circulation of air is prevented. The "inlets" referred to are holes in the base of each neutron-absorbing tube. Cooling fluid (water when the pool is full, air when it is empty) can enter through these holes and, as it rises convectively, extract decay heat from the spent fuel assemblies. The most likely cause of blocked inlets is the presence of residual water at the base of each neutron-absorbing tube. Thus, less-than-total loss of water from the pool will be more significant than total loss.

The dashed curve in Figure Q.1 shows the effect of including oxidation effects in the calculations. The oxidation reaction between air and the zirconium fuel cladding is exothermic and proceeds rapidly at temperatures above 1000°C. Thus, as will be seen from Figure Q.1, a "run-away" reaction can occur.

A similar reaction will occur between steam and zirconium; this reaction is also exothermic and can also "run away" at temperatures above 1000°C. In the event of partial water loss, this reaction will occur rather than the air-zirconium reaction.

The calculations behind Figure Q.1 assume one-year-discharged fuel. Clearly, recently discharged fuel will be most susceptible to the initiation of an exothermic reaction. However, once such a reaction is initiated, the resultant heat can bring the cladding of adjacent fuel assemblies up to the ignition temperature. By this means, a zirconium "fire" can spread through the pool, involving older fuel assemblies as well.

This "fire" would be characterized by glowing of the cladding rather than by flames. Gradually, the cladding would become weakened and many of the UO_2 pellets would become exposed. Volatile radionuclides, particularly cesium, would be released from those pellets to the atmosphere within the pool building.

If the reaction were between zirconium and steam, then hydrogen would be evolved in significant quantities. A hydrogen explosion in the pool building could then occur, leading to a breach in that building. Such a breach would create a direct path whereby radionuclides in the building atmosphere could reach the outside environment.

Further analytic, and some empirical, work is required, so that our understanding of this accident scenario may be improved. For example, the calculations behind Figure Q.1 are not sufficiently sophisticated. However, enough is known to substantiate the description given above[Q.2].

At this juncture, the reader may reasonably ask: "Under what circumstances will there be total or partial loss of water from a spent fuel pool?"

At some PWRs (and even more BWRs), the design of the pool is such that it is easy to envisage the pool becoming totally or partially drained due to sabotage or earthquake damage, or via an accident during refuelling. At the Sizewell PWR, total drainage will not occur during such incidents unless the pool wall or base is breached, which would require a quite determined act of sabotage or a major earthquake. There is no opening in the pool walls below the top of the fuel assemblies[Q.3].

For Sizewell, a scenario of greater interest is a reactor accident which interrupts cooling of the pool water and prevents access to the pool building by repair teams. In that event, the pool water will evaporate and eventually expose the fuel assemblies. In a typical case, the pool water would begin to boil about 2 days after cooling was lost. The pool would boil dry after a further 19 days[Q.4].

After a serious reactor accident, radiation fields near the pool building could prevent human access for times of this order. Access could be prevented even if the reactor accident did not lead to a very large atmospheric release. For example, an accident involving melt-through of the basemat, without above-ground containment failure, might lead to intense radiation fields in the immediate vicinity of the containment building, due to radioactive steam and gases rising from the ground.

Via this scenario, a reactor accident could lead to a release of a significant fraction (perhaps tens of percent) of the cesium in the spent fuel. The total cesium release from the combined reactor and pool accidents could then be substantially greater than the release from the reactor accident alone. The area of land which would become unsuitable for habitation would increase correspondingly.

3. On-Site Cask Storage

There are several methods of on-site spent fuel storage which are less dangerous than high-density pool storage. Perhaps the most interesting of these methods is dry storage in casks.

Figure Q.2 shows a West German cask storage concept. In this plan, for the Wurgassen plant, a group of 40 casks would be located in a building on the plant site. Each cask would hold four spent fuel assemblies. More buildings could be added as needed.

With this concept, no power or water supplies are required for the cooling of the spent fuel. Human intervention is confined to routine oversight. The casks, if properly designed and built, will be safe against most events except severe fires, acts of war, or determined sabotage. Moreover, casks can be added as new storage capacity is required, thus avoiding the high initial cost associated with some other storage concepts.

In the US, three companies have submitted information on their respective cask designs to the Nuclear Regulatory Commission (NRC). One of these companies (Combustion Engineering) has proposed a cask which can hold 24 PWR spent fuel assemblies. Also, the US Department of Energy intends to demonstrate cask storage in cooperation with utilities in Virginia and North Carolina, and with the Tennessee Valley Authority[Q.5, Q.6].

4. Transport Incidents

During transport, spent fuel will be held in heavy shipping casks. In normal circumstances, transport poses little risk. However, there are a number of abnormal circumstances which could lead to a public health risk.

A severe impact could lead to deformation or rupture of the cask, and damage to the fuel assemblies. Also, a release path from the cask interior to the environment could be created by cask rupture, or by damage to cask seals or valves. Noble gases and volatile fission products (particularly cesium) could be released. If the impact were accompanied by fire, greater release would be expected.

The Greater London Council (GLC) will be presenting evidence on this matter at the Sizewell Inquiry, drawing upon work by the consulting firm Technica. As part of this effort, the GLC has commissioned the UK National Radiological Protection Board (NRPB) to estimate the public health effects of various possible releases arising from a rail accident at Willesden Junction (in London).

The NRPB has published some of the results of their investigation. Their assumed release fractions are shown in the first column of Table Q.1. The assumed accident is an impact followed by a 2-hour fire at about 1000°C. In the mean outcome, NRPB predicts 2 fatal cancers, and in the 99th percentile case (only 1% of outcomes would be worse) they predict 14 fatal cancers[Q.7].

A detailed study of spent fuel transportation has recently been published by the Council on Economic Priorities (CEP), an independent organization based in New York[Q.8]. This CEP study finds that higher release fractions than those assumed by the NRPB are credible. The second column of Table Q.1 shows release fractions which CEP find credible for impact plus a fire leading to an internal cask temperature of 1000°C. It should be noted that short-cooled (say, 1 year) fuel is assumed.

Sabotage is also a real possibility. A study by Sandia Laboratories shows that explosives, particularly shaped charges, could breach both truck and rail casks[Q.9]. For truck-mounted casks, Sandia estimates that fractions of the spent fuel mass from 0% to 100% could be displaced from the cask, and fractions from 0.7% to 100% could be scattered as solid particles. Up to 0.2% (baseline estimate: 0.07%) of the solid contents could be released as an aerosol. The third column of Table Q.1 summarizes Sandia's release estimates (for gaseous or aerosol release).

Sandia's release estimates do not consider the effect of fire as part of a sabotage event. Once the cask has been breached, air can reach its interior and oxidize the zirconium fuel cladding and the UO_2 fuel pellets themselves. Thus, in view of the release fractions which CEP finds credible for impact/fire scenarios, higher release fractions than the Sandia numbers seem credible for sabotage/fire scenarios. The fourth column in Table Q.1 shows tentative estimates of release fractions for such scenarios.

5. Notes and Sources

(Q.1) CEGB, Sizewell'B' PWR Pre-Construction Safety Report, April 1982, Chapter 13.

(Q.2) The author, with colleagues, is currently investigating this subject. For an earlier account of the author's understanding, see: (i) "Potential Accidents and Their Effects," Report of the Gorleben International Review, 1979, Chapter 3 [Note: This document is available (in German) from the government of Lower Saxony, West Germany, and also (in English) from the Political Ecology Research Group, Oxford, UK.]; and (ii) G. Thompson, Testimony Concerning the Proposed Increase of Spent Fuel Storage Capacity at Prairie Island Nuclear Plant, presented to the Minnesota Energy Agency, June 1980.

(Q.3) One could, however, envisage a sabotage scenario involving siphoning water from the pool through one of the water return lines (which terminate at the bottom of the pool).

(Q.4) The assumptions behind this calculation are:

water volume:	1500 m^3
decay heat:	2 MW
pool temperature before cooling loss:	50°C
mean water depth:	5m

These parameters are roughly characteristic of an almost-filled pool, mid-way between refuellings. For further information, see ref (Q.1).

(Q.5) US Nuclear Regulatory Commission, 1982 Annual Report, June 1983, pp. 64-65.

(Q.6) US Department of Energy, Department of Energy to Negotiate Cooperative Agreements for Spent Fuel Storage Demonstrations, press release, 5 October 1983.

(Q.7) R.H. Clarke and K.B. Shaw, "Consequences of Release of Activity during Irradiated Fuel Transport," Proceedings of the Conference on the Urban Transportation of Irradiated Fuel, Connaught Rooms, London, April 1983, MacMillan (in press).

(Q.8) M. Resnikoff, Study Director, The Next Nuclear Gamble: Transportation and Storage of Nuclear Waste, Council on Economic Priorities, 1983.

(Q.9) N.C. Finley et al., Transportation of Radionuclides in Urban Environs: Draft Environmental Assessment, NUREG/CR-0743, July 1980.

Table Q.1

Various Estimates of Radionuclide Release Fractions (percent) for Incidents Involving Spent Fuel Transport Casks

	NRPB's[a] Impact and Fire Scenario	CEP's[b] Impact and Fire Scenario	Sandia[c] Sabotage Scenario	Sabotage/Fire[d] Scenario
Noble Gases	30	?	10-25	10-100
Cesium	0.03	10	0.02-0.2	1-30
Ruthenium[e]	0.03	1	0.02-0.2	?
Tellurium	1×10^{-4}	10	0.02-0.2	?
Other Nuclides	1×10^{-4}[f]	?	0.02-0.2	?

[Notes and Sources on next page.]

Notes and Sources for Table Q.1

(a) See ref (Q.7).

(b) See ref (Q.8), Chapter VI.

(c) See ref (Q.9), Section 5.

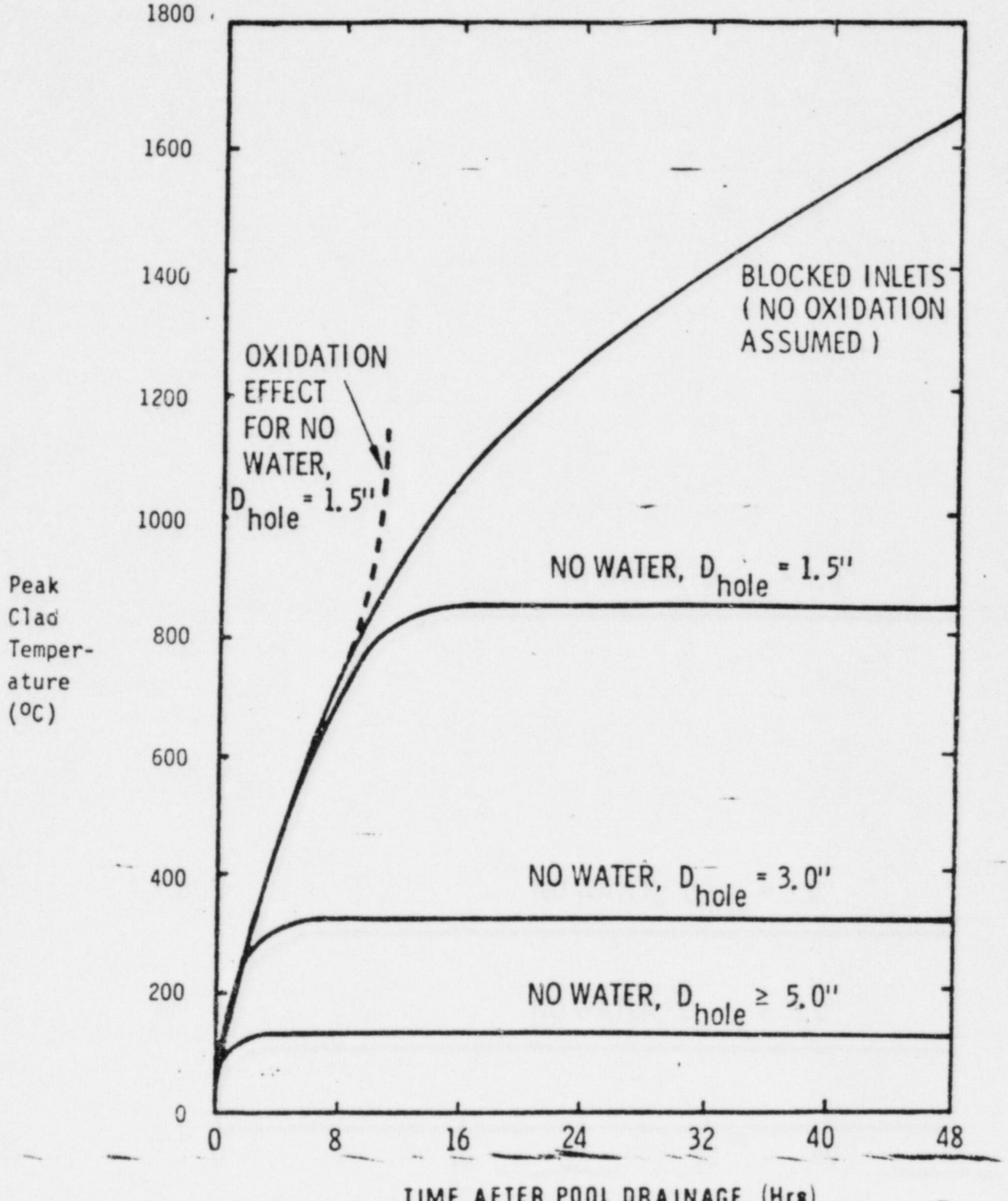
(d) Tentative estimates by author--see text.

(e) Ruthenium is highly volatile in the tetroxide form.

(f) Except cobalt, for which a release fraction of 0.25% was assumed.

Figure Q.1

Estimated Cladding Temperature Following Loss of Water from a Spent Fuel Pool Containing PWR Spent Fuel in Compact Racks



(Notes and Sources on next page)

Notes and Sources for Figure Q.1

(a) This figure adapted from Fig 26 of A.S. Benjamin et al, Spent Fuel Heatup following Loss of Water During Storage, US Nuclear Regulatory Commission report NUREG/CR-0649, March 1979

(b) The spent fuel is assumed to be placed in upright cylindrical canisters which are open at the top and which have a hole of diameter D at the bottom. It is assumed that fluid flow cannot occur in the spaces between the canisters.

(c) The pool will contain batches of spent fuel of varying ages. In this instance, the fuel is assumed to be aged one year after discharge from the reactor.

(d) The cases marked "NO WATER" refer to complete loss of water from the pool. Decay heat is then removed primarily by upward convection of air. Larger D leads to lower clad temperature.

(e) The case marked "BLOCKED INLETS" results from partial loss of water, so that upward convection of air is inhibited. Decay heat must then be removed by upward and downward radiation and by evaporation of the residual water.

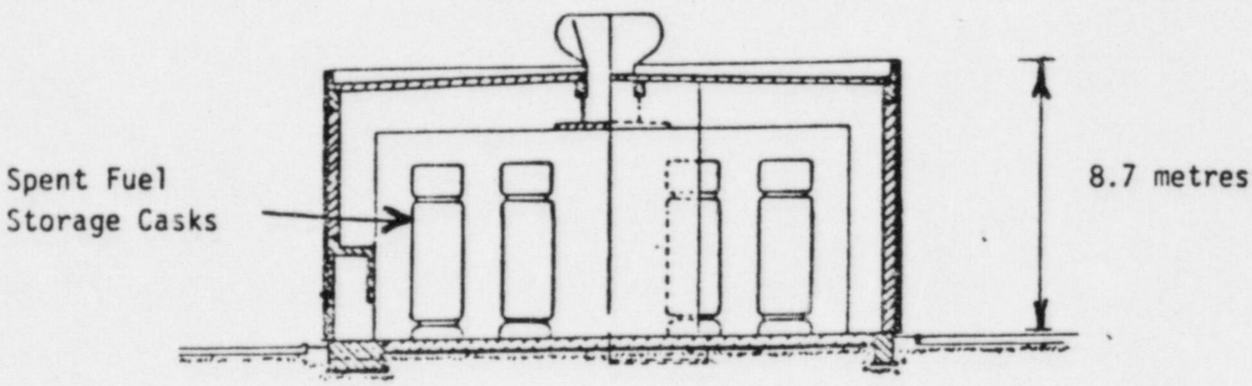
(f) The dashed line indicates the effect of including cladding oxidation in the calculation.

Figure Q.2

Concept for Interim Storage of Spent Fuel at

Reactor Sites Using Dry Casks

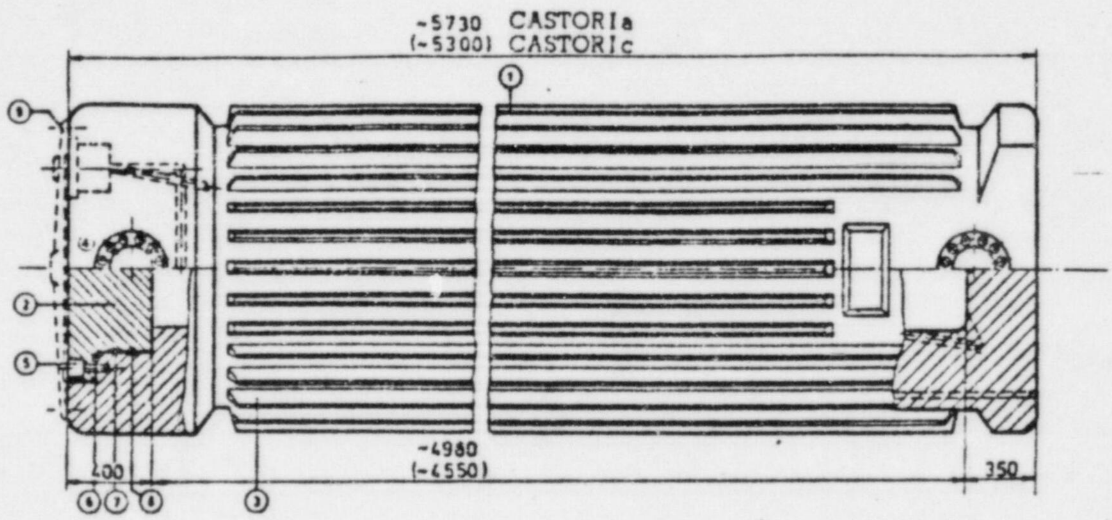
(i) The Storage Building



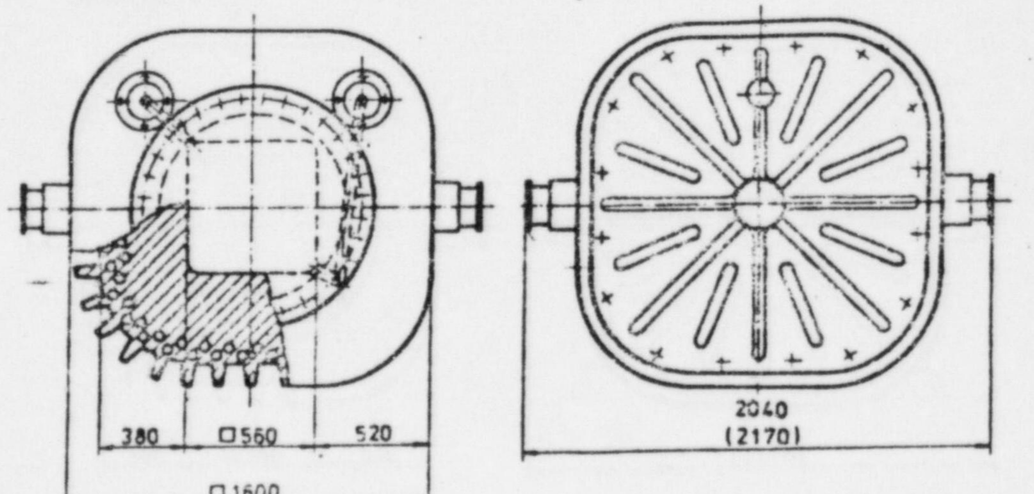
(continued on next page)

Figure Q.2 (continued)

(ii) The Cask
(dimensions in millimetres)



CASTORIa



(Notes and Sources on next page)

Notes and Sources for Figure Q.2

(a) The drawing of a storage building is from documents prepared in 1979 by Preussen-Elektra of Hannover, for their license application for interim storage at the Wurgassen plant in West Germany.

(b) The drawing of a Castor cask is from Transportbehalterlager, Die trockene Lagerung von ausgedienten Brennelementen, Deutsche Gesellschaft fur Wiederaufarbeitung (undated).

(c) In this Preussen-Elektra concept, each building would hold 40 casks.

(d) The Castor la cask shown is intended for 4 PWR fuel assemblies (2.1 MTHM).

APPENDIX B

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,
Consultant,
Center for Energy and
Environmental Studies,
Princeton University,
Princeton, NJ 08544

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 <u>one</u> 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.

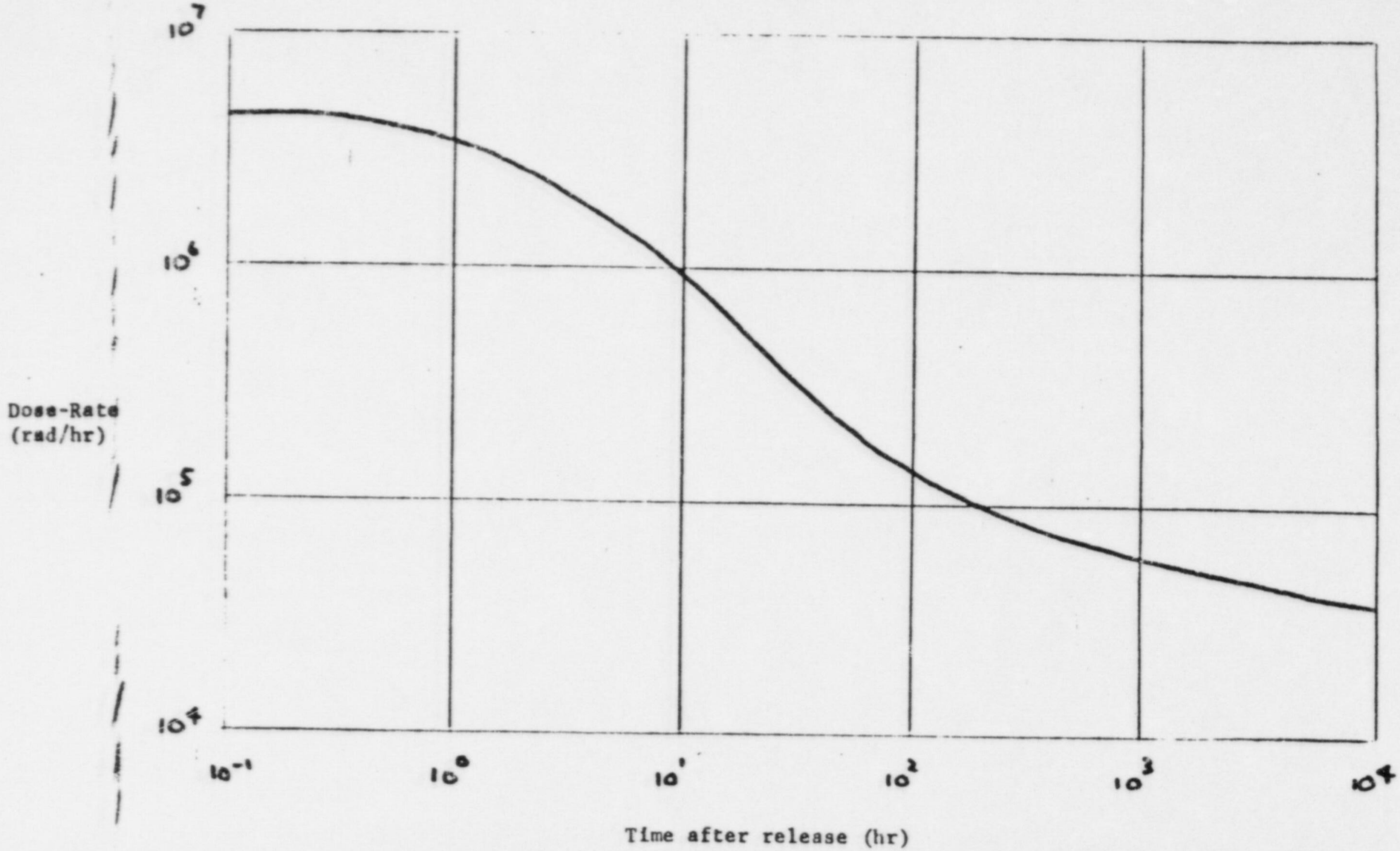


Figure 1: Gamma Dose-Rate Inside Containment Building following Loss-of-Coolant Accident
Source: Final Safety Analysis Report, Salem Units 1 and 2, Public Service Electric and Gas Co.

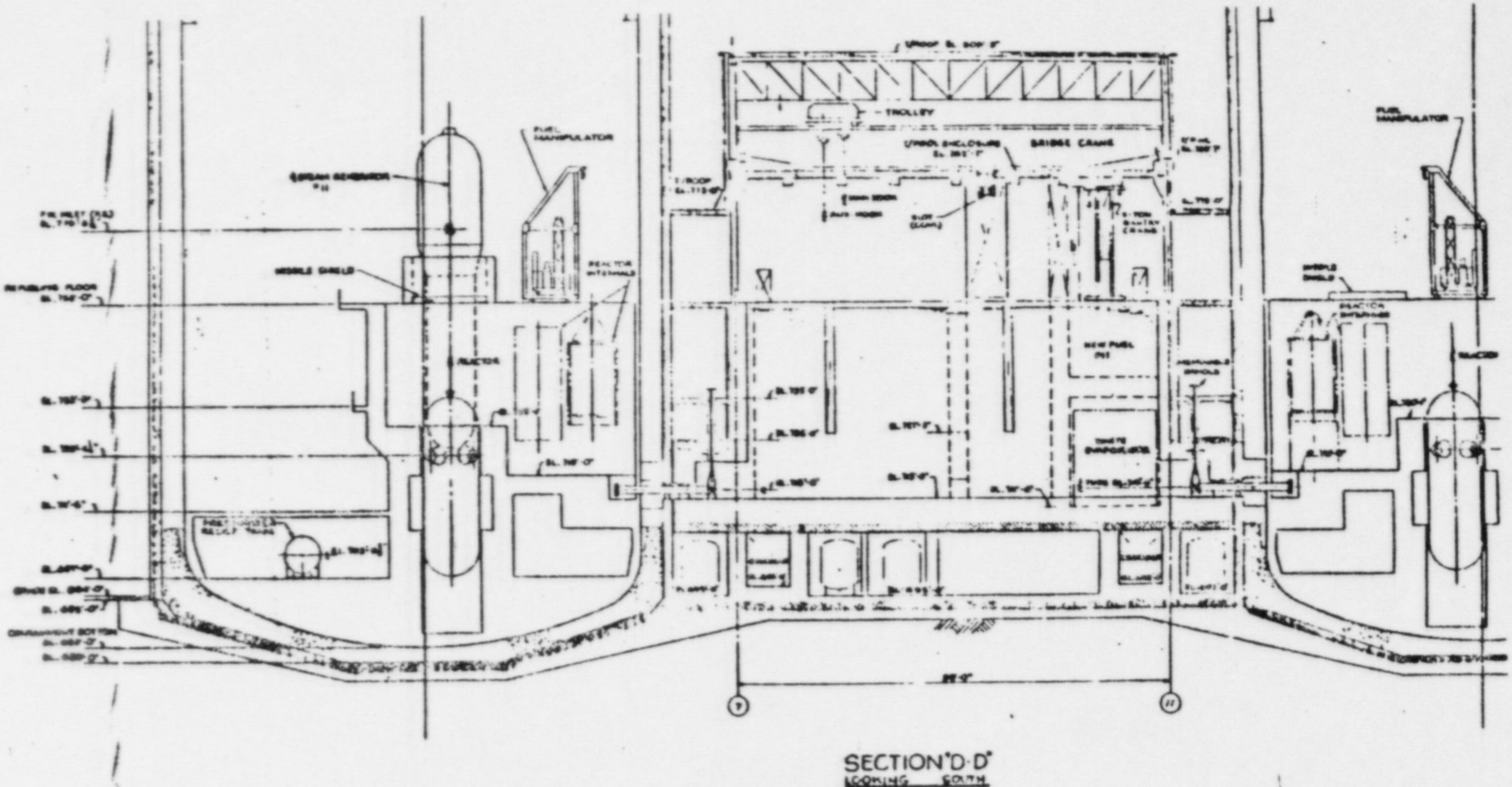


Figure 2 : Cross-Section of Prairie Island Plant Showing Spent Fuel Pools

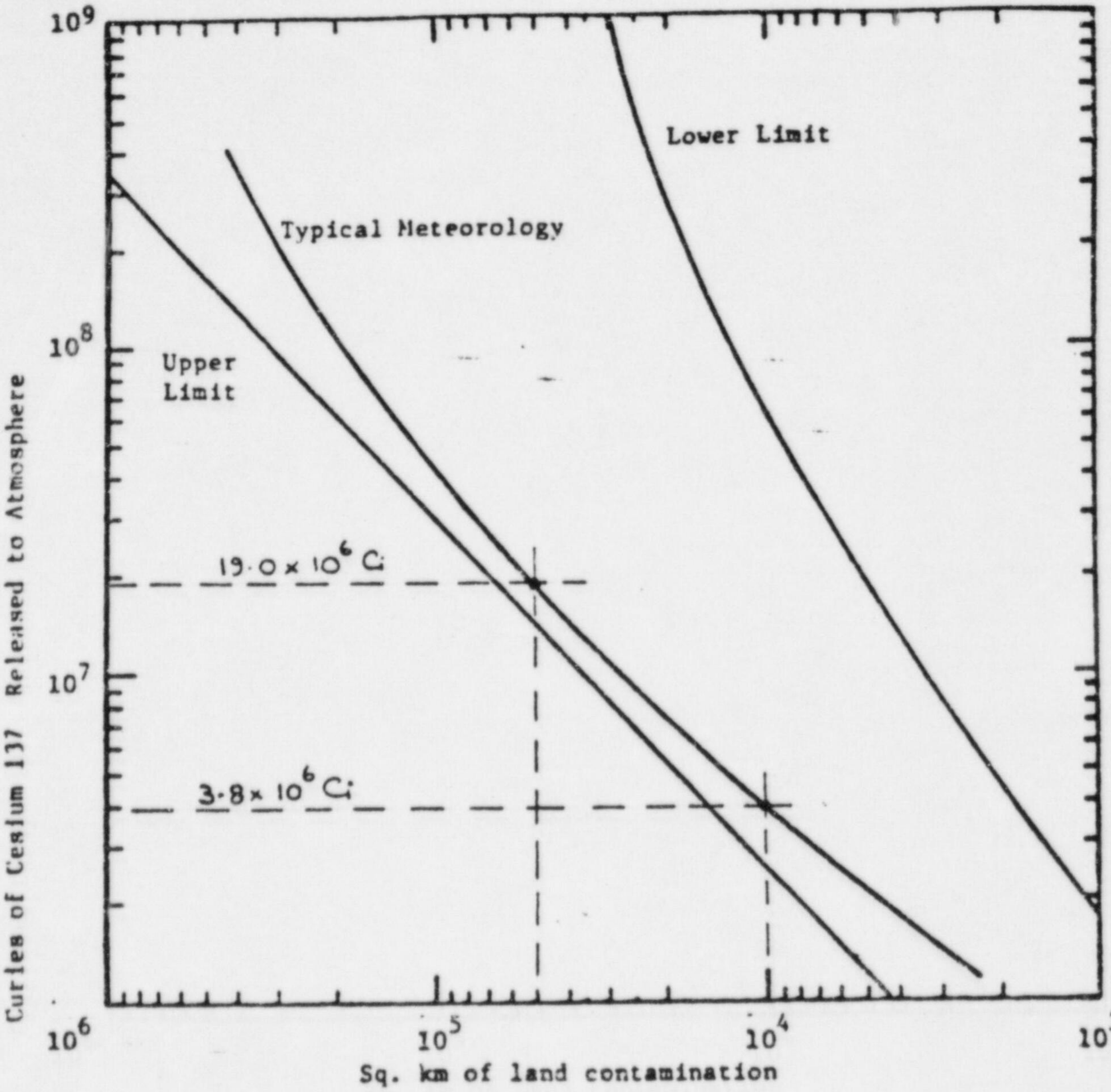


Figure 3 : Area of Land Contaminated by Atmospheric Release of Cesium 137

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 120°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	← 48 te HM
60 days	40	
1 yr	80	
.	.	
.	.	
15 yrs	80	
Total :	1360	← 544 te HM

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^{o}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^{o}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 (^{o}K)

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 = 6.2×10^{-4} for a PWR, leading to the following results :

cladding temperature (^{o}C)	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°K to just above its melting point (3030°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

Resume
for
Gordon Thompson

June 1986

Professional Expertise

Consulting scientist on energy, environment, and international security issues.

Education

* PhD in Applied Mathematics, Oxford University, 1973.
* BE in Mechanical Engineering, University of New South Wales, Sydney, Australia, 1967.
* BS in Mathematics and Physics, University of New South Wales, 1966.

Current Appointments

* Executive Director, Institute for Resource & Security Studies (IRSS), Cambridge, MA.
* Coordinator, Proliferation Reform Project (an IRSS project).
* Treasurer, Center for Atomic Radiation Studies, Acton, MA.
* Member, Board of Directors, Political Ecology Research Group, Oxford, UK.
* Member, Board of Directors, New Century Policies Educational Programs Inc, Cambridge, MA.
* Member, Advisory Board, Gruppe Okologie, Hannover, FRG.

Consulting Experience (selected)

* Lakes Environmental Association, Bridgton, ME, 1986 : analysis of federal regulations for disposal of radioactive waste.
* Three Mile Island Public Health Fund, Philadelphia, PA, 1983-present : studies related to the Three Mile Island nuclear plant.
* Attorney General, Commonwealth of Massachusetts, Boston, MA, 1984-present : analyses of the safety of the Seabrook nuclear plant.
* Union of Concerned Scientists, Cambridge, MA, 1980-1985 : studies on energy demand and supply, nuclear arms control, and the safety of nuclear installations.
* Conservation Law Foundation of New England, Boston, MA, 1985 : preparation of testimony on cogeneration potential at the Maine facilities of

Great Northern Paper Company.

* Town & Country Planning Association, London, UK, 1982-1984 : coordination and conduct of a study on safety and radioactive waste implications of the proposed Sizewell nuclear plant.
* US Environmental Protection Agency, Washington, DC, 1980-1981 : assessment of the cleanup of Three Mile Island Unit 2 nuclear plant.
* Center for Energy & Environmental Studies, Princeton University, Princeton, NJ, 1979-1980 : studies on the potentials of various renewable energy sources.
* Government of Lower Saxony, Hannover, FRG, 1978-1979 : coordination and conduct of studies on safety aspects of the proposed Gorleben nuclear fuel center.

<u>Other Experience (selected)</u>

* Co-leadership (with Paul Walker) of a study group on nuclear weapons proliferation, Institute of Politics, Harvard University, 1981.
* Foundation (with others) of an ecological political movement in Oxford, UK, which contested the 1979 Parliamentary election.
* Conduct of cross-examination and presentation of evidence, on behalf of the Political Ecology Research Group, at the 1977 Public Inquiry into proposed expansion of the reprocessing plant at Windscale, UK.
* Conduct of research on plasma theory (while a PhD candidate), as an associate staff member, Culham Laboratory, UK Atomic Energy Authority, 1969-1973.
* Service as a design engineer on coal plants, New South Wales Electricity Commission, Sydney, Australia, 1968.

<u>Publications (selected)</u>

* <u>Nuclear-Weapon-Free Zones : A Survey of Treaties and Proposals</u> (edited with David Pitt), Croom Helm Ltd, Beckenham, UK, forthcoming.
* <u>The Source Term Debate : A Report by the Union of Concerned Scientists</u> (written with Steven Sholly), January 1986, Union of Concerned Scientists, Cambridge, MA.
* "Checks on the spread" (a review of three books on nuclear proliferation), <u>Nature</u>, 14 November 1985, pp 127-128.
* Editing of <u>Perspectives on Proliferation</u>, Volume I, August 1985, published by the Proliferation Reform Project, Institute for Resource and Security Studies, Cambridge, MA.
* "A Turning Point for the NPT ?", <u>ADIU Report</u>, Nov/Dec 1984, pp 1-4,

University of Sussex, Brighton, UK.

* "Energy Economics", in J Dennis (ed), The Nuclear Almanac, Addison-Wesley, Reading, MA, 1984.
* "The Genesis of Nuclear Power", in J Tirman (ed), The Militarization of High Technology, Ballinger, Cambridge, MA, 1984.
* A Second Chance : New Hampshire's Electricity Future as a Model for the Nation (written with Linzee Weld), Union of Concerned Scientists, Cambridge, MA, 1983.
* Safety and Waste Management Implications of the Sizewell PWR (prepared with the help of 6 consultants), a report to the Town & Country Planning Association, London, UK, 1983.
* Utility-Scale Electrical Storage in the USA : The Prospects of Pumped Hydro, Compressed Air, and Batteries, Princeton University report PU/CEES #120, 1981.
* The Prospects for Wind and Wave Power in North America, Princeton University report PU/CEES # 117, 1981.
* Hydroelectric Power in the USA : Evolving to Meet New Needs, Princeton University report PU/CEES # 115, 1981.
* Editing and part authorship of "Potential Accidents & Their Effects", Chapter III of Report of the Gorleben International Review, published in German by the Government of Lower Saxony, FRG, 1979 -- Chapter III available in English from the Political Ecology Research Group, Oxford, UK.
* A Study of the Consequences to the Public of a Severe Accident at a Commercial FBR located at Kalkar, West Germany, Political Ecology Research Group report RR-1, 1978.

Expert Testimony (selected)

* International Physicians for the Prevention of Nuclear War, 6th Annual Congress, Koln, FRG, 1986 : Relationships between nuclear power and the threat of nuclear war.
* Maine Land Use Regulation Commission, 1985 : Cogeneration potential at facilities of Great Northern Paper Company.
* Interfaith Hearings on Nuclear Issues, Toronto, Ontario, 1984 : Options for Canada's nuclear trade and Canada's involvement in nuclear arms control.
* Sizewell Public Inquiry, UK, 1984 : Safety and radioactive waste implications of the proposed Sizewell nuclear plant.
* New Hampshire Public Utilities Commission, 1983 : Electricity demand and supply options for New Hampshire.
* Atomic Safety & Licensing Board, Dockets 50-247-SP & 50-286-SP, US Nuclear Regulatory Commission, 1983 Use of filtered venting at the Indian

Point nuclear plants.

* US National Advisory Committee on Oceans and Atmosphere, 1982 : Implications of ocean disposal of radioactive waste.
* Environmental & Energy Study Conference, US Congress, 1982 : Implications of radioactive waste management.

Miscellaneous

* Australian citizen.
* Married, one child.
* Resident of USA, 1979 to present; of UK, 1969-1979.
* Extensive experience of public speaking before professional and lay audiences.
* Author of numerous newspaper, newsletter, and magazine articles and book reviews.
* Has received many interviews from print and electronic media.

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APPENDIX D

Declaration of the State Government,

Lower Saxony, West Germany

by

Minister - President
Dr. Ernst Albrecht

May 16, 1979

Concerning the proposed nuclear fuel centre
at Gorleben

(English Translation)

In November 1976 I had the honor to receive, in the presence of the fraction chairmen of CDU, SPD and FDP, the Federal Ministers Maihofer, Friderichs and Matthöfer. The members of the Federal Government informed the State Government about the planned integrated fuel cycle center ("Entsorgungszentrum") and requested the immediate selection of a preliminary site for this center.

On February 22, 1977, the State Government announced their readiness to examine applications for the construction of an Entsorgungszentrum on the Gorleben site. Independent of the examination as prescribed for the procedure according to atomic law, however, the question whether an integrated Entsorgunszentrum was fundamentally realizable from the viewpoint of safety technology was to be clarified first. The safety of the population, the State Government stated, has to have priority over all other considerations.

On March 31, 1977, the DWK (Deutsche Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen mbH, German Association for Reprocessing of Nuclear Fuels Ltd.) submitted the application for the licensing of the construction of the nuclear Entsorgunszentrum. The application for the construction of a final deposit for radioactive wastes on the Gorleben site was submitted on July 28, 1977 by the Physikalisch-Technische Bundesanstalt (PTB, Physical-Technological Federal Institute).

The State Government has carefully examined the problems which arise in connection with the construction of an Entsorgungszentrum.

For this purpose, they relied on the council of numerous highly qualified experts. The reactor safety commission and the commission for radiological protection issued a statement. In March 1979, the topic was the subject of an intense debate between more than 60 international scientists (Gorleben-Symposium). After these careful investigations, the Lower Saxony State Government issues the following preliminary statement:

A. On the safety of the plant:

The State Government has arrived at the conclusion that the final disposal of radioactive wastes in a suitable salt dome entails no risk for the present generation as well as for those of the immediate future. For later generations, the risk is small compared to other risks of life.

Because of their plasticity, the salt domes in Northern Germany have endured for over 100 million years without being touched in their core. Several glaciations and geo-historical catastrophies, such as the separation of the american continent from the european continent, could not harm them. Nevertheless, not every salt dome and not every part of a salt dome is equally suited for final disposal. The suitability has to be examined by careful investigations (drillings, geophysical investigations, opening of shafts). Scientific and technological methods are available for this purpose.

By an adequate cooling-down period of the radioactive wastes and by storing them in a sufficiently large volume, it can be guaranteed that the stability of the salt dome will not be decreased by the heat released by the high-activity waste materials.

A risk for future generations would arise only if in the course of the centuries the knowledge about the disposal of radioactive materials would be lost and later generations, uninformed about the final disposal, would attempt to open up the salt dome by mining.

Although in this case, however, it is to be pointed out that the toxicity of final deposits with wastes from reprocessing will be drastically reduced after 500 to 1000 years and will then be comparable to the toxicity of natural deposits of mercury-, lead- and uranium-ores.

More problematical, however, are the facilities connected to the reprocessing plant. The question of the safety of these facilities has to be posed with the local population, the workers and employees of the Entsorgungszentrum, as well as the population of the Federal Republic of Germany and its neighbours in view.

1. The safety of the local population

Here, we have to distinguish between the normal operation of the nuclear Entsorgungszentrum (NEZ) and the results of possible incidents

a) Normal operation

Like all nuclear facilities, the nuclear Entsorgungszentrum will release certain amounts of radioactivity to the environment. According to the regulations of the radiological protection ordinance, the yearly whole-body-dose for each single person living in the immediate vicinity of the NEZ must not exceed 30 mrem (rem is a unit for the radiation exposure of single persons. 1 rem = 1000 mrem) [illegible] via air and water. Beside this, corresponding limits for the maximal permissible radiation exposure of individual organs such as the thyroid are prescribed.

The State Government has come to the conclusion that it is possible to stay considerably below these maximal values. They would require the operator to stay below a dose of ten mrem per year.

The compliance with this limit would be controled by permanent monitoring of emmissions (in particular at the off-gas stacks) as well as by permanent monitoring of immissions in the surroundings of the NE

If necessary, the State Government would not hesitate to temporarily shut down the plant to guarantee that the maximal yearly dose is not exceeded.

Scientists agree that each radiation exposure in addition to the natural exposure can have health effects.

The risk entailed by the above-mentioned maximal dose of ten mrem per year and person, however, is far smaller than other risks of life with which our population is acquainted. The natural radiation exposure in the Federal Republic is ca. 110 mrem per year. The use of x-rays for diagnostic purposes leads, in the population average to ca. 50 mrem per year and person.

In the Federal Republic of Germany, about 25 persons per year and per 10 000 inhabitants die of cancer. This is about 1/6 of all deaths The operation of the nuclear Entsorgungszentrum would increase this cancer risk for the local population from 25 to 25,01, if each person would be exposed to 10 mrem per year (estimation of the UN-committee for the investigation of the effects of atomic radiation). Due to the rapid reduction of radiation exposure with increasing distance, the majority of the local population will be subjected to a considerably lower risk.

If the calculation is based on the maximal values used by the nuclear energy critics at the Gorleben-Symposium, the risk is increased from 25 to 25,06.

b) Incidents in the interior of the plant

Incidents inside the chemical factory proper (part project 2), i.e. in the reprocessing plant itself, can be controled. This also applies to the retention technology which controls the release of radioactive materials to the environment.

The State Government thinks that it can guarantee that incidents inside the reprocessing plant itself will not lead to a radiation exposure of the population above the legal limits. This, however, will necessitate cost-intensive safety precautions.

The State Government recognizes that the stores, which contain over 95 % of the radioactive plant inventory, constitute a special hazard potential. 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;
- high-activity wastes are, in normal operation, not stored in liquid form and that buffer tanks, if such are necessary, are made inherently safe.

2. The safety of workers and employees

The State Government could convince themselves that the operational safety in the planned nuclear Entsorgungszentrum can be at least as good as in other industrial facilities.

All large industrial facilities contain certain risks. According to present experience, the radiation exposure (whole-body dose) of the personnel working in the control area of the plant will not exceed 1,5 rem per year. The risk given thereby, or in other words the reduction of the average life expectancy resulting from this exposure is of about equal size as the reduction of the life expectancy of steel workers and significantly smaller than the risk which professional drivers, fishermen and miners working underground take upon themselves when they are practicing their profession.

Incidents can in the short term lead to radiation exposures inside the plant which are higher than normal. In so far this has no immediate health effects it will have to be decided in each single case whether the persons concerned will have to be removed temporarily or permanently from the control area of the plant.

The permanent health control of the whole personnel is important for the State Government. Whole-body monitoring permits a reliable determination of the radiation exposure of the individual workers and employees.

3. The safety of the population in the Federal Republic of Germany and the neighbouring countries

If the requirements of the State Government (see A. 1. b) are fulfilled, the population living further away from the plant will not be influenced by the normal operation of the facility and by inciden taking place inside the plant.

There remain, however, two risks which can not be excluded with certainty.

One is the risk of the impact of war. One can assume that particularly if the geographic location is considered - the parties engaged in the conflict will try to avoid a destruction of the plan which would entail the risk of a release of a fraction of the radioactive potential. Furthermore, the State Government would shut down the plant in case of war. An impact due to war nevertheless cannot be completely excluded.

In order to exclude, in this case, risks, which exceed the average risk level already created by the war, the State Government require in addition to the modifications formulated in 1. b) the developmen of a concept to store radioactive substances which could be dispersed underground in case of war.

A further risk is the possibility of a theft of plutonium for terrorist purposes.

The State Government is convinced that the plutonium store can be constructed and secured in a manner which renders access of terrorists from outside impossible.

Theft of plutonium by members of the personal, however, can not be excluded to the same extent. It is for the Federal Government to know whether they want to carry the political risk this constitutes.

The following summary can be given: On the assumption that the concept of DWK will be subject to essential modifications, it is possible to construct a nuclear Entsorgungszentrum in such a manner that population and personnel will not be exposed to higher risks in their life than they are by other industrial and technological facilities which the population is already accustomed to. This safety-technological answer, however, is not sufficient. Even if a reprocessing plant, in principle, can be built and operated so safely that it does not lead to unacceptable risks for the population, the question remains of whether the construction of such a plant is absolutely necessary and whether it appears to be politically realizable.

B. The political and energy-policy aspects

Today, 14 nuclear power plants are already in operation in the Federal Republic of Germany and nine more are being built at the moment. In any case, spent fuel from those plants has to be taken care of (the plants have to be "entsorgt"). Furthermore, it is the opinion of the Federal Government and the State Government that the energy demand of the future can only be covered in a satisfactory manner with a contribution from nuclear energy.

It would be wrong to consider the construction of an integrated Entsorgungszentrum as the only solution of the "Entsorgungs"-question. It has been established that long-term intermediate storage of spent fuel elements for several decades is technically possible in a save manner. Regarding final disposal, there is, in principle, the choice between final disposal after reprocessing and final disposal without reprocessing.

The direct final disposal of spent fuel elements after a longer cooling-off period is possible in principle even if development work is still required for the technical realization. Direct final disposal avoids the problems of reprocessing. On the other hand, it means that wastes with a high content of plutonium have to be deposited for a long time in salt domes or in other geologic formations. The State Government is convinced that, in principle, the wastes can be stored in a safe manner; however, the remain toxic for a significantly longer period than a final deposit after reprocessing.

The advantages of reprocessing for waste management and waste disposal should not be regarded as small; however, it can be stated that the real advantages of reprocessing will only materialize in combination with the fast breeder. Indeed, this combination permits a 60-fold utilization of the nuclear fuel. Thereby, the Federal Republic of Germany would be able to significantly reduce its dependence from other countries, an important aspect in the long-term perspective of a world in which a bitter fight for these scarce energy reserves cannot be excluded. This is a decision, however, which can only be taken in years and after the testing of the breeder at Kalkar.

There is no necessity to begin the construction of a reprocessing plant today as long as the decision on the fast breeder is open. This consideration gains particular weight in connection with the question of the political requirements for a realization of a nuclear Entsorgungszentrum.

It cannot be doubted that during the last years the fear of the risks of nuclear installations has grown in large parts of our population.

In spite of it being legally possible - with good reason - , the State Government does not consider it right to build a reprocessing plant as long as it has not been possible to convince large parts of the population of the necessity and safety-technological acceptability of the plant. In contrast to many other decisions, this is not a question of competing interests; it is a question of a judging health risks. Therefore, the opinion of the immediately concerned population carries particular weight.

Whether it will be possible to convince the population will depend not last on the position the parties take. It is not possible to expect the population to gain confidence in the nuclear Entsorgungszentrum if the politically responsible hold different opinions in this matter. Exactly that, however, is the case today. Leading politicians, organizations on State and district level as well as working groups of SPD and FPD have spoken against the reprocessing plant. Others go still further and take position against nuclear energy in general. It is a task of foremost political importance to create a clear situation in this field.

The Lower Saxony State Government cannot and does not want to force energy-political decisions upon the Federal Government. It is their duty, however, to point out to the Federal Government that the political preconditions for the construction of a reprocessing plant are not given at the moment.

C. Summary

Although a nuclear Entsorgungszentrum is, in principle, realizable from the viewpoint of safety-technology, the Lower Saxony State Government recommends the Federal Government to not further persue the project of reprocessing.

The new "Entsorgungs"-concept should be decided instead without delay The basic features of this concept can be described as follows:

- Immediate installations of inherently safe long-term intermediate stores for the "Entsorgung" of the nuclear power plan
- Pushing of research and development activities for the safe final disposal of radioactive waste.
- Deep drillings and, if the results are positive, opening up of a mine in the Gorleben salt dome. In case the drillings should lead to negative results, investigation of other fina disposal sites.
- Decision of the most appropriate form of treatment and final disposal of radioactive waste only after clarity on the energ political future has been reached.

This concept permits safe "Entsorgung". It does not foreclose any options for the future. It limits the risks connected to "Entsorgung" to a minimum.

Depending on whether the Federal Republic of Germany will in the future opt for light water reactors, for the high-temperature reactor or for the fast breeder, the question of reprocessing can then be taken up again. The long-term intermediate storage guarantees that no nuclear fuel gets lost.

The Lower Saxony State Government is willing to participate in the realization of such a concept. Concretely spoken, this means the willingness to install a long-term intermediate storage facility, to realize the final disposal of low- and intermediate-activity wastes in salt domes in Lower Saxony, after the procedures requir by law have been executed, and to push the mining investigations for the final disposal of high-activity materials.

A part of this task, e.g. the construction of long-term intermedia stores, can also be taken over by other Federal States. The State Government would consider it wrong to let those states out of the duty. We are, however, aware of the fact that Lower Saxony has a particular responsibility due to its geographic characteristics, and we will act according to this responsibility.

REPORT OF THE GORLEBEN INTERNATIONAL REVIEW

CHAPTER 3

POTENTIAL ACCIDENTS AND THEIR EFFECTS

*Submitted to the Government of Lower Saxony,
West Germany, in [illegible].

(*Submitted in the German language -
this version completed September 1979)

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THE STORY OF THE GORLEBEN INTERNATIONAL REVIEW

A consortium of West German electric utilities wished to build at Gorleben (in the State of Lower Saxony, West Germany) a nuclear fuel centre encompassing spent fuel storage, reprocessing, waste disposal and fuel fabrication.

The Lower Saxony State Government (as licensing authority) responded to public unease by commissioning a review of the project by 20 international critical scientists. The resulting report (Chapter 3 herewith) was submitted in March 1979 and subjected to a semi-public examination during 28 March - 2 April, 1979, attended throughout by the state governor (Dr. Albrecht) and several of his cabinet. Five critical German scientists and approximately 35 scientists favourable to the project participated, in addition to the 20 international critics.

On 16 May 1979, Dr. Albrecht announced that the project would not now be licensed and that future re-application would not be considered without changes in design (copy of Albrecht's statement follows).

Gorleben International Review Report: Chapter 3

Potential Accidents and their Effects

(a) Table of Contents

(b) Chairman's Introduction

This chapter represents the work of a GIR sub-group consisting of:

J. Beyea
Y. Lenoir
G. Rochlin
G. Thompson (Chairman)

The authorship of each section is shown at the head of that section.

All matters raised have been discussed within the sub-group, with other members of the GIR panel, with the co-ordinator and with others. The responsibility for each section is, however, that of the stated author.

Gorleben International Review Report: Chapter 3

3.8 Loss of Cooling to Spent Fuel Storage (SFS) Ponds

(This section by G. Thompson)

3.8.1 Summary

Studies have been conducted in BRD and UK which show that SFS ponds have the potential for catastrophic release if their cooling systems are interrupted for more than a few days.

As for the similar situation of HAW tanks (see section 3.7.1), the SB does not consider this possibility and RSK/SSK and TÜV accept that omission.

We undertake an illustrative study of the consequences of cooling loss for the DWK concept of TP1.

It is found that pond water will boil away and expose the fuel elements after times of 90-250 hrs. depending on pond heat load. Fuel cladding will then reach temperatures in excess of 1000°C and a steam-zircalloy reaction will follow. This reaction will generate hydrogen and an explosion leading to breach of the pond building can be expected. The heat of reaction will result in a substantial release of activity to the atmosphere. 600 million curies of Ru106 and 300 million curies of Cs137 could be released.

This scenario requires nothing more than neglect. Alternative initiating events such as explosion, aircrash or relatively minor acts of war (see sections 3.4 and 3.5) could initiate a similar release. The timescale before release might be very short in such cases if cracking of the pond walls leads to water loss.

3.3.2 Introduction

This section serves the same function as section 3.7 on loss of services to HAW tanks. As stated there, our analysis provides a brief illustration of the kind of accident study which DWK would have included had they written a complete SB.

3.3.3 Description of SFS Ponds (from the SB).

3.8.3.1 Layout

Six ponds are provided, each with a capacity of 500 te.

Pond dimensions are approx:

Length 16.3 m

Width 9.2 m

(Water)
Depth 14.0 m

Fuel is vertically racked in the base of the pond in a 7 x 4 horizontal array of 2 m square racks.

The layout of a rack for PWR fuel is shown in Fig. 3.8-1. It will be noted that each rack will accept 49 fuel elements. Each element is surrounded by a 3 mm thick boron steel case, to prevent criticality problems which might arise from the close packing adopted.

The ponds are housed in two groups of 3 within parallel and inter-connected halls. Each hall provides an air chamber above the ponds of approx. dimensions:

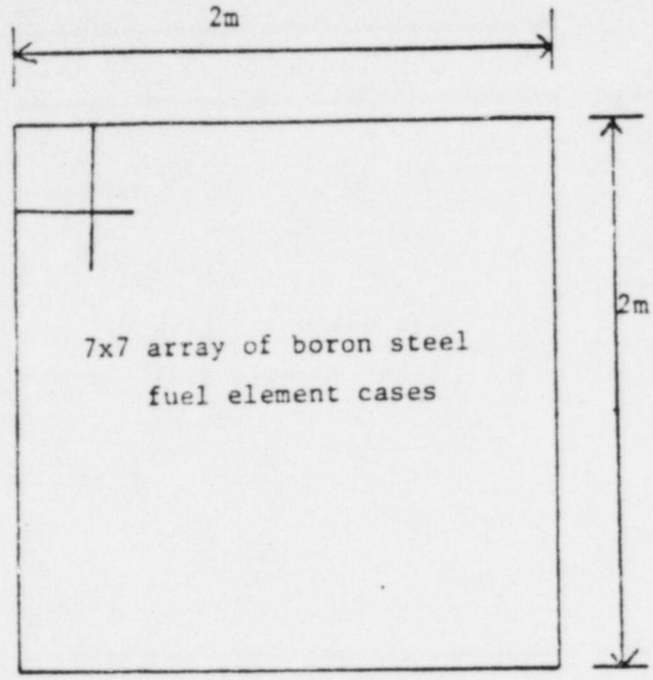
Length 90 m

Width 28 m

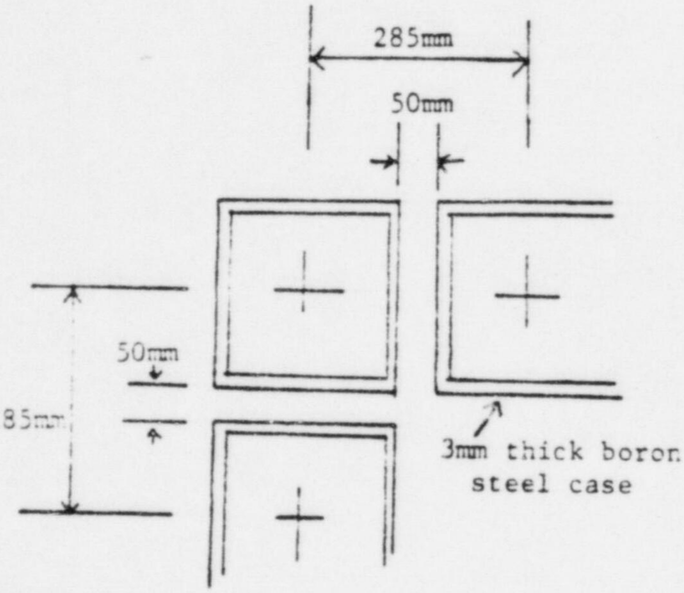
Height 20 m

Fig. 3.8-1 Schematic Views of Layout of PWR Spent Fuel Rack (from the SB)

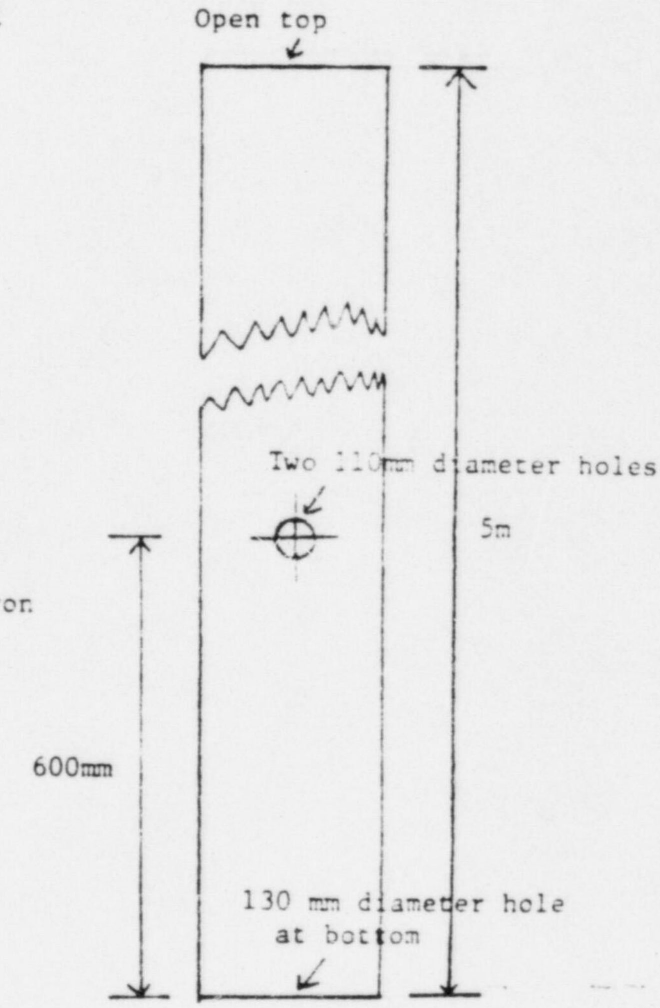
Note: All dimensions approximate only.



(a) Horizontal cross section of rack



(b) Detail of horizontal cross-section



(c) Vertical view of boron steel-case

The halls are provided with inner walls approx. 0.4 m thick. The inner walls are separated from the outer walls by an air-space of 2.5 m or at least its equivalent (in terms of thermal insulation) by way of service ducts etc. The outer walls are approx. 2 m thick. The base of the building is approx. 2.5 m thick. The pond walls are approx. 1.5 m thick and the base of each pond is approx. 2 m thick.

It is clear that this arrangement offers a certain level of security against external influences. It also provides effective thermal insulation of the ponds from the environment.

3.8.3.2. Mode of Operation

The minimum age of the fuel after discharge from the reactor is to be 180 days.

The ponds will normally be loaded with 40% BWR fuel and 60% PWR fuel although other variations are possible.

The maximum heat to be removed from a single pond is to be 13.25 MW, and from all 6 ponds 48 MW. Normal operating temperature is not more than 40°c.

3.8.4 Previous discussion of Cooling Loss

3.8.4.1 Discussion in the SB

The SB considers cooling loss for a few hours only. Results are provided (Table 1.5.2.4 - 1) for the rate of temperature rise of pond water in the event of cooling loss. This rate varies from 2.1 to 5.5 $^{\circ}$ c/hr for the cases considered.

No justification is provided for the restriction of calculations to such a limited period.

3.8.4.2 Findings of RSK/SSK

RSK/SSK state[1] that "cooling of spent fuel elements must be guaranteed in the event of all conceivable accidents." They accept that such a guarantee is provided by application of the "single-fault with repair" criterion.

They quote an investigation by DWK of the effectiveness of natural cooling given the present design of pond building. With a 2 atmosphere overpressure of steam within the building, only 6% of decay heat can be removed by natural processes (convection, conduction, phase change).

RSK/SSK state that "An inherently safe system (natural circulation) is considered to be unfeasible without loss of protection against external factors."

3.8.4.3 Dialogue of GIR and DWK

In discussions between GIR and DWK,[2] the latter repeated the RSK/SSK assertion that natural cooling could not be combined with protection against external events.

DWK stated that cooling loss for more than a short period could be ruled out and that application of the "single fault with repair" criterion and spatial separation of redundant parts of the cooling system would provide sufficient guarantee of this. The guidelines of the Federal Ministry of Interior regarding reactor safety were referred to as justification for such a view.

3.8.4.4 Dialogue of GIR and TÜV

During discussions(3) between GIR and TÜV, the latter stated that they would analyse cooling loss for a period of 10 hours only, as repairs could be made by that time.

TÜV do not consider alternative designs, they simply analyse the project as submitted.

3.8.4.5 Evidence at the Windscale Inquiry

During that inquiry, BNFL undertook calculations, at the instruction of the presiding Inspector, on the time-scale of events following loss of cooling to SFS ponds.

The results presented(4) included estimates of time-scales for the boiling away of pond water and of the maximum temperature attained by fuel elements.

3.8.4.6 Work at the Institute for Reactor Safety, Köln

The IRS have produced a study(5) which includes calculations of the time-scale of boiling away of the pond water and calculations of the doses received from a possible release following such water loss.

3.8.5 The Need to Consider Cooling Loss

The need to consider loss of services has been discussed in section 3.7.5, in connection with HAW tanks. DWK have calculated in the SB the rate of temperature rise of SFS pond water in the event of cooling loss, but have not considered the boiling period.

3.8.6 Events Following Cooling Loss

3.8.6.1 Heating up of Pond Water to Boiling Point

DWK have provided (Table 1.5.2.4 - 1 of the SB) figures for the rate of rise of temperature under adiabatic conditions. We reproduce those figures and also show the time required to rise from normal operating temperature (40^{o}c) to boiling point (110^{o}c assumed as an average). The results are shown in Table 3.8-1.

Table 3.8-1

Heating up of SFS Pond Water to Boiling Point

Heat Load of Pond (MW)	Rate of Temperature Increase (^{o}c/hr)	Time from 40^{o}c to 110^{o}c (hrs)
13.25	5.5	12.7
9.9	4.2	16.7
7.9	3.3	21.2
6.5	2.7	25.9
5.6	2.4	29.2
4.9	2.1	33.3

3.8.6.2 Boiling Away of Pond Water

If we take the same heat loads as in Table 3.8-1 and again assume adiabatic conditions, the rate of boiling can be calculated. We assume that the phase change requires 2.23 MJ/kg (i.e. boiling at 1.4 bar).

We calculate the time taken to expose the top of the fuel elements (9 m depth boiled away) and to expose one half of the active length of a PWR element(11.4 m depth boiled away). The results are shown in Table 3.8-2. Pond dimensions are discussed in section 3.8.3.1, above.

Table 3.8-2

Boiling Away of SFS Pond Water

Heat Load of Pond (MW)	Time to Expose Top of Fuel Elements (hrs)	Time to Expose ½ of active length of PWR elements (hrs)
13.25	63.4	80.3
9.9	84.8	107.4
7.9	106.3	134.6
6.5	129.2	163.7
5.6	150.0	190.0
4.9	171.4	217.1

Note

Times shown are from the beginning of the boiling period.

From Tables 3.8-1 and 3.8-2 we see that the cumulative time from loss of cooling to exposure of ½ the active length of the fuel elements varies from 93 hours to 250 hours. It is of interest that BNFL presented evidence (6) to the Windscale Inquiry on the effect of a temperature of 100°c on the concrete walls of a SFS pond. BNFL state that, after several days, some cracking would occur, leading to leakage. Thus the longer time-scales shown here might be reduced.

3.8.6.3 Heat Transfer to the Environment Before and During Boiling

In sections 3.8.6.1 and 3.8.6.2 we have assumed adiabatic conditions. This appears reasonable in view of the arrangement of the pond building as discussed in section 3.8.3.1, above.

Additionally, the DWK calculations mentioned above in section 3.8.4.2 show that conditions will be approximately adiabatic.

3.8.6.4 Heat Transfer from the Exposed Fuel Elements

3.8.6.4.1 Introduction

Heat transfer at this stage is complicated and more detailed analysis is required. For our illustrative analysis, we assume that each 500 te capacity pond contains 1000 PWR elements with the following characteristics:

- fuel rods in 16 x 16 array
- 236 rods in place
- element envelope cross-section is 21cm x 21cm
- 500 kg U or fission products per element
- rod outer diameter 1 cm
- cladding thickness 1mm
- active length 3.9 m
- inactive length 0.4 m (top), 0.7 (bottom)
- interior of boron steel case is 23 cm x 23 cm

For the assumed situation, the heat output of an average fuel rod can be calculated, as shown in Table 3.8-3.

Table 3.8-3

Heat Output of Average Fuel Rod

Pond Heat Load (MW)	Rod Heat Output (W)
13.25	56.1
9.9	41.9
7.9	33.5
6.5	27.5
5.6	23.7
4.9	20.8

We will consider each heat transfer mechanism separately and then summarize our findings.

3.8.6.4.2 Heat Transfer by Conduction

The area of cross-section of the cladding in each rod is 2.83×10^{-5} m^2. The cross-sectional area of the fuel pellet (neglecting gap) is 5.03×10^{-5} m^2.

Respective thermal conductivities are taken from ref (7):

For zircalloy cladding: 17.3 W/mK

For fuel pellet: 1.99 W/mK

We can now take the rod heat outputs of Table 3.8-3 and approximately calculate temperature gradients along the rod, (assumed equal for cladding and fuel pellet) if conduction is the only heat transfer mechanism. The results are shown in Table 3.8-4.

Table 3.8-4

Approximate Temperature gradient along Fuel Rods for Heat Transfer by Conduction Only

Rod Heat Output (W)	Temperature Gradient (^{o}c/m)
56.1	2.44×10^{4}
41.9	1.82×10^{4}
33.5	1.46×10^{4}
27.5	1.20×10^{4}
23.7	1.03×10^{4}
20.8	9.04×10^{3}

Note

In this simplified model the heat source in each m length is assumed concentrated at the middle of that length. Such a model gains some validity from the fact that decay heat is greater near the middle of the rod.

It is clear that fuel rod integrity will not be maintained in the above situation (melting point of zircalloy is 1800^{o}c).

3.8.6.4.3 <u>Heat Transfer by Radiation</u>

There are two pathways for such radiation:

(i) Along the narrow passages between fuel rods.

(ii) Laterally (via a combination of reflection and azimuthal conduction in cladding) to the boron steel case surrounding the elements.

The size of passage available betwen the fuel rods is indicated by the dimension of that cylinder which can be fitted between the rods (axes parallel). Such a cylinder would have a diameter of 0.98 cm. Thus the ratio of length to diameter of passage per m length of fuel element is of order 10^2. Transfer by this route will be small.

We note from Reilly et al.[8] that the length over which longitudinal thermal radiation might be important is about 10 cm.

Regarding lateral transfer to the boron steel case, we note that Naitoh et al.[9] have conducted theoretical and experimental work on the similar problem of transfer to a BWR channel box. Their experiment (in air) shows a temperature drop from the central rods to the channel box of approx. 200°c (our estimate, as channel box temperature is not given). Although the heat output per te is higher in their situation, the additional rods in our (PWR) situation will probably compensate, making the two situations roughly comparable.

If the heat transferred to the boron steel case were to be transferred longitudinally by conduction, then an analysis such as that leading to Table 3.8-4 shows (assuming thermal conductivity of the steel to be 50 W/mK) that a pond heat load of 13.25 MW corresponds (approximately) to a temperature gradient of 2.46×10^4 °c/m.

Heat can be transferred longitudinally by radiation in the space between boron steel cases, which are approx. 5 cm apart. The argument of Reilly et al. quoted above suggests that such longitudinal transfer will be important over a length of approx. 50 cm only. In any event, the rack design shown in the SB has substantial restrictions at the top of each gap.

It will be noted that absorption by water vapour will reduce radiative transfer, although the effect is relatively small. From Welty(10) we can see that the emissivity of water vapour in our situation is not likely to exceed 0.1.

It is of interest to note the temperature attained by the tops of the fuel elements in the event that they are required to radiate away all the heat reaching them. We assume:

- half of pond heat load is radiated away from the upper surface of the elements
- the radiating surface is black
- the radiating area is that of the plane of the top of the racks (8 m x 14 m)
- incoming radiation is negligible

A pond heat load of 13.25 MW then corresponds to a temperature of the radiating surface of 738^{o}c. A heat load of 7.9 MW corresponds to 615^{o}c.

It is clear from the above that heat transfer by the combined processes of radiation and conduction will result in cladding temperatures at the inner regions of the fuel elements well in excess of 1000^{o}c. The significance of this figure we will see later.

3.8.6.4.4 Heat Transfer by Natural Convection

Let us consider the situation where part of the fuel element is exposed and part is covered by water. It will be seen from Figure 3.8-1 that convection must then occur in vertical channels which are closed at the bottom by water.

A related situation is discussed by Bonilla[11] who presents results of experiments in air involving two uniformly heated parallel vertical plates 1.3 m wide and 1.8 m high, confined at the sides and bottom and with spacing down to 7.5 cm.

The experiment described is comparable to the situation of convection in the gaps between the boron steel cases. These gaps will be the most important sites for convection. We assume that the cross-sectional area receiving heat from each fuel element is 0.026 m^2. We take from Welty[12] the physical properties of air at 1000^{o}K (the highest temperature for which properties are tabulated) and find that for a pond heat load of 13.25 MW the wall temperature is of the order of 10^{4} ^{o}c. Although more complete data is required (Grashof number declines rapidly with increasing temperature) it is clear that wall temperatures will exceed failure points. The outcome is quite similar if natural convection of steam is considered. We also note, as in section 3.8.6.4.3 above, that the rack design shown in the SB includes substantial restrictions at the top of each gap.

3.8.6.4.5 Forced Convection by Steam

While the fuel elements are partly exposed, steam will be generated at their lower ends and this steam will become superheated as it rises past the exposed upper ends. The superheating process for the steam is also a cooling process for the elements.

An interesting feature of this situation is that the temperature of steam leaving the top of the fuel element depends only on the fraction of the element exposed and not on the pond heat load. If we take the average specific heat of steam from 100°c to 800°c (at 1 bar) as 2.1 kJ/kgK and the latent heat of boiling of water (at 1 bar) as 2260 kJ/kg, we have:

$$T = \left(\frac{2260}{2.1} \times \frac{e}{(1 - e)}\right) + 100$$

where: T(°c) is temperature of steam leaving the upper part of the fuel element

e is the fraction of active length of fuel exposed.

Some results are shown in Table 3.8-5.

Note: Those who remark the singularity of the above equation for e = 1 will recall that the present discussion treats each heat transfer mechanism separately. Thus temperature will be stabilized by other heat transfer processes, but at temperatures well in excess of 1000°c.

Table 3.8-5

Temperature of Steam Leaving Fuel Element

Exposed Fraction of active length, e	Steam Temperature, T (^{o}c)
0.3	561
0.4	817
0.5	1175
0.6	1713
0.7	2609

It will of course be noted that cladding temperature will exceed steam temperature.

3.8.6.4.6 Summary of Findings on Heat Transfer from Exposed Elements

It will be clear from sections 3.8.6.4.2 to 3.8.6.4.5 that the dominant heat transfer mechanism is that of forced convection by steam. This mechanism offers no advantage to lower pond heat loads and leads to cladding temperatures well in excess of 1000^{o}c.

3.8.6.5 Initiation of Steam-Zircalloy Reaction

It will be noted from Lewis[7] that the reaction:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

becomes significant for cladding temperature above 1000^{o}c.

If an adequate supply of steam is available, the reaction follows a rate law:

$$\frac{dr}{dt} = - \frac{3.97 \times 10^{-5}}{(r_o - r)} \quad \exp\left(- \frac{22889}{T}\right)$$

where:

r = radius of reacting interface (m)

r_o = initial radius (m)

T = interface temperature (^{o}K)

t = time (sec)

The first part of this function accounts for the inhibiting effect of the accumulating oxide layer.

If we consider an early stage of the reaction, when $r/r_o = 0.99$ and assume a cladding temperature of 1175°c (taken from Table 3.8-5 for an exposure of ½ of each fuel element), we find (for $r_o = 5$ mm):

- $\frac{dr}{dt} = 6.5 \times 10^{-6}$ m/min
- mass burned per m length of fuel rod (density 6.55 te/m^3) = 2.2×10^{-5} kg/sec
- heat output per m length of fuel rod (6.53 MJ/kg) = 0.15 KW 3
- heat output per pond (1000 elements, 236 rods per element, ½ of active length exposed) = 67 MW
- Zr consumption per pond = 10.1 kg/s
- H_2 evolution per pond = 0.44 kg/S
- H_2O consumption per pond = 4.0 kg/S

It will be noted that this reaction is steam-limited on the basis of steam generated by decay heat (2.9 kg/S for 13.25 MW pond heat load and 50% exposure of elements). However, a portion of the 67 MW additional heat output will enter the pond water by radiation and thereby generate larger quantities of steam.

If all the cladding in a rod were to react, the heat released would be 6.0 MJ. If this heat were all transferred to the 2.1 kg of fuel pellets (assuming specific heat of the pellets to be 300 J/kgK), their temperature would rise to 9300 °c. It will be noted from Lewis[7] that fuel melting occurs at 2800°c and vaporization at 3300°c. If the reaction is sufficiently fast (and therefore more nearly adiabatic) then some fuel melting will occur.

The progress of this reaction requires more detailed study in view of the interactive effects of heat transfer, structural behavior and reaction rate. We note from Lewis[7] that if more than about 18% of cladding is oxidized then the cladding becomes susceptible to fragmentation from thermal shock. In any event, as pointed out by BNFL,[4] cladding will rupture under internal pressure at about 700°c.

3.8.6.6 Release of Activity from Fuel

In view of the substantial energy release which can occur from steam-zircalloy reaction it must be supposed that radionuclides will be released from the spent fuel.

For this illustrative study we make the (probably conservative) assumption that release fractions will be the same as those we assume for HAW residue (see section 3.7.9), namely:

Ru, Cs	90%
Sr, Ce, Pm	5%
Pu	1%

In order to estimate the radiological effect of such a release, we assume a reference case in which 1500 te of 1-yr-discharged fuel and 1500 te of 2-yr-discharged fuel is stored.

Once a steam-zircalloy reaction has commenced in one pond, it must be assumed that similar reactions will be initiated in the other 5 ponds. The ponds are interconnected and their walls subject to failure as a result of the heat output of fire in an adjacent pond. In any event, it may be that the ponds will be similarly loaded and will therefore behave in parallel.

We assume release of activity from the complete inventory of 3000 te. The activity per te of fuel is taken from Table 3.7-1 of Section 3.7 on HAW. Releases of our selected nuclides are then as shown in Table 3.8-6.

Table 3.8-6

Release of Selected Radionuclides from SFS Ponds for Reference Case

Nuclide	Pond Inventory (Ci)	Release (Ci)	Release (kg)
SR90	2.3×10^8	1.2×10^7	86
RU106	6.2×10^8	5.6×10^8	165
CS137	3.3×10^8	3.0×10^8	3450
CE144	9.8×10^8	4.9×10^7	15
PM147	2.3×10^8	1.2×10^7	13
PU238	8.4×10^6	8.4×10^4	5
PU239	9.9×10^5	9.9×10^3	158

Note

Assumptions are as discussed in section 3.8.6.6 of text.

3.8.6.7 Escape of Activity from Pond Building

The heat released within the building from steam-zircalloy reaction may be well in excess of that from decay heat (400 MW in the situation discussed in section 3.8.6.5, compared with 48 MW design heat load).

The resultant increase in temperature of the walls of the building will reduce their strength. We note from Callahan et al.[13] that concrete loses strength with increasing temperature, complete loss of strength occurring by 1050°K.

Substantial H_2 production will occur. If the steam-zircalloy reaction proceeds to completion then 6 ponds (3000 te fuel) will generate 57 te of H_2 (from 510 te H_2O).

Taking the air space within the building to have a volume of 1.0×10^5 m^3, we find that the lower flammability level (4% H_2 by volume) is reached following the evolution of 0.31 te H_2.

It is clear that the combination of weakened concrete and a H_2 explosion allows an entirely plausible assumption of a substantial breach in the pond building. All of the releases shown in Table 3.8-6 will then pass directly into the atmosphere.

3.8.7 Radiological Effects of Release to Atmosphere

Section 3.13 of this chapter (by Beyea) discusses such effects.

The plume issuing from the breach in the pond building will rise as a result of heat generated from steam-zircalloy reaction. The effective release height of 300 m assumed in section 3.13 seems a reasonable approximation. We note that the only result sensitive to release height is the range of 1-yr. 10 thousand rem lung dose.

Release duration is uncertain due to the lack of detailed knowledge on the progress of the steam-zircalloy reaction, as discussed in Section 3.8.6.5. Section 3.13 assumes a 3-hr duration.

The nature and scale of effects is discussed in section 3.13, where they are seen to be considerable.

3.8.8 Radiological Effects of Release to Groundwater

It may be that the steam-zircalloy reaction will conclude at a time when some water remains in the ponds. Such water, or that entering from the surrounding ground-water via cracks in the pond and building walls, will leach activity from the remains of the spent fuel. Thus, activity could be carried into ground-water. Activity deposited on the ground near the site will also enter local ground-water.

As outlined in section 3.12, the characteristics of the Gorleben site are such that activity entering the ground-water is likely to appear in the Elbe at times less than 100 yrs.

3.8.9 Alternative Circumstances Leading to Similar Release

As discussed in section 3.7.13 in connection with HAW tanks, we note that other circumstances than the simple neglect assumed here can lead to release.

Alternative scenarios include those featuring aircrash, explosion, sabotage and acts of war.

A feature of particular importance for SFS ponds is that severe cracking of pond walls and resultant leakage of water will almost immediately lead to the initiation of steam-zircalloy reaction. There may well be insufficient time for emergency measures (e.g. flooding the pond building).

3.8.10 Acknowledgements

The efforts of Peter Taylor are noted in bringing the hazard potential of SFS ponds to the attention of the UK public and this author, in the context of the Windscale Inquiry.

Frank von Hippel has contributed by discussion on the fate of exposed fuel elements.

3.8.11 References

(1) State of Consultations of RSK and SSK concerning Gorleben at 15 February 1978, translated by GIR.

(2) Meeting of 27 September 1978 between GIR and DWK.

(3) Meeting of 6 January 1979 between GIR and TÜV.

(4) See the proceedings of the Windscale Public Enquiry 1977, particularly BNFL documents nos. 299 and 309 submitted in evidence.

(5) Bachner et al., Report No. 290 of Institut für Reaktorsicherheit, Köln. (August 1976)

(6) BNFL document no. 223 submitted in evidence to the Windscale Public Inquiry 1977.

(7) E.E. Lewis, "Nuclear Power Reactor Safety" published by John Wiley & Sons (1977).

(8) J. T. Reilly et al. "The Effects of Thermal Radiation on the Temperature Distribution in Fuel Rod Arrays" Nuclear Engineering and Design Vol. 48 pp. 340-351 (1978).

(9) M. Naitoh et al., "Analysis of Radiant Heat Transfer in a BWR Fuel Assembly" Nuclear Engineering and Design Vol. 44, pp. 315-321 (1977).

(10) J. R. Welty, "Engineering Heat Transfer" published by John Wiley (1978). Fig. 6.39 refers to emissivity of water vapour.

(11) C. F. Bonilla, Chapter 2 (Heat Transfer) of Reactor Handbook 2nd edition ed. S. McLain and J. H. Martens published by Interscience (1964).

(12) as (10). Tables A-3 refer to physical properties of gases.

(13) J. P. Callahan et al. "Uniaxial Compressive Strengths of Concrete for Temperatures Reaching 1033K" Nuclear Engineering and Design, Vol. 45, pp. 439-448 (1978).