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February 5, 1987

Mr. Edward Throm
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RE: FIN A-3786

Dear Ed:

I have enclosed four copies of the revised draft to the report entitled, "Beyond Design-Basis Accidents in Spent Fuel Pools (Generic Issue 82)." We have incorporated your comments on the rough draft. The multidisciplinary nature of this effort required input from several organizations within BNL. The management review is still ongoing.

This report satisfies the milestone for the draft report. We will complete the formal NUREG/CR after receiving the NRC review.

Sincerely,

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NUREG/CR-
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BEYOND DESIGN-BASIS ACCIDENTS IN SPENT FUEL POOLS (GENERIC ISSUE 82)

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ABSTRACT

This investigation has provided an integrated assessment of the risk of beyond design basis accidents in spent fuel pools for two surrogate plants (a PWR and BWR). The investigation included an assessment of initiating frequency, analyses of the accident progression including the fission product releases and health consequences. The estimated health consequences were found to be about 12 person-rem/Ry and 130 person-rem/Ry for the BWR and PWR plants, respectively. These estimated risk results are comparable to the estimated risk posed by severe core damage accidents and appear to warrant further attention. However, the uncertainty in this estimate is large (greater than a factor of 10) and plant specific features may change the results considerably.

Preventive and mitigative measures have been evaluated qualitatively. It is suggested that for plants with similar risk potential to the two surrogate plants, the one measure which is likely to be effective in reducing risk is utilization of low density storage racks for recently discharged fuel. However, before such preventive measures are implemented a complete plant specific risk assessment for pool related accidents should be performed including a structural fragility analysis of the pool itself.

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BEYOND DESIGN-BASIS ACCIDENTS IN SPENT FUEL POOLS (GENERIC ISSUE 82)

SUMMARY

S.1 INTRODUCTION

Generic Safety Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools," was assigned MEDIUM priority in November 1983.[1] In its prioritization, the NRC staff took account of two factors that had not been considered in earlier risk assessments:[2]

1. Spent fuel is currently being stored rather than shipped for reprocessing or repository disposal, resulting in much larger inventories of spent assemblies in reactor fuel basins than had previously been anticipated; and,

2. A theoretical model[3,4] suggested the possibility of catastrophic Zircaloy fire, propagating from assembly to assembly in the event of complete drainage of water from the pool.

S.1.1 Previous Investigations

The Reactor Safety Study[2] (which did not take account of the two factors above) concluded that the risks associated with spent fuel storage were extremely small in comparison with accidents associated with the reactor core. That conclusion was based on design and operational features of the storage pools which made the loss of water inventory highly unlikely.

Subsequent to the Reactor Safety Study, A.S. Benjamin et al.[3,4] investigated the heatup of spent fuel following drainage of the pool. A computer code, SFUEL, was developed to analyze thermal-hydraulic phenomena occurring when storage racks and spent assemblies become exposed to air.

Calculations with SFUEL indicated that, for some storage configurations and decay times, the Zircaloy cladding could reach temperatures at which the

exothermic oxidation would become self-sustaining with resultant destruction of the cladding and fission product release. The possibility of propagation to adjacent assemblies (i.e., the cladding would catch fire and burn at a hot enough temperature to heat neighboring fuel assemblies to the ignition point) was also identified. In such cases, the entire inventory of stored fuel could become involved. Cladding fires of this type could occur at temperatures well below the melting point of the UO_2 fuel. The cladding ignition point is about 900°C compared to the fuel melting point of 2880°C.

S.1.2 Related Events

There is no case on record of a significant loss of water inventory from a domestic, commercial spent fuel storage pool. However, one recent incident occurred at the Haddam Neck reactor that raised concern about the possibility of a partial draindown of a storage pool as a result of seal failure in the refueling cavity at a time when the transfer tube gates to the pool were open, or when transfer of a spent fuel assembly was in progress.[5]

The Haddam Neck incident occurred during preparations for refueling. An inflatable seal bridging the annulus between the reactor vessel flange and the reactor cavity bearing plate extruded into the gap, allowing 200,000 gallons of borated water to drain out of the refueling cavity into the lower levels of the containment building in about 20 minutes. Gates to the transfer tube and the fuel storage pool were in the closed position, so no water drained from the pool.[6]

More recently a pneumatic seal failure in the Hatch spent fuel basin which released approximately 141, 000 gallons of water resulted in a drop in water level in the pool of about five feet.[7]

S.1.3 Report Objective

The objective of this report is to provide an integral assessment of the risk potential of beyond design basis accidents in spent fuel pools. The risks are defined in terms of

- the probabilities of various initiating events that might compromise the structural integrity of the pool or its cooling capability,
- the probability of a system failure, given an initiating event,
- fuel failure mechanisms, given a system failure,
- potential radionuclide releases, and
- consequences of a specified release.

This study generally follows the logic of a typical probabilistic risk analysis (PRA); however, because of the relatively limited number of potential accident sequences, the analyses are greatly simplified.

S.1.4 Spent Fuel Storage Pool Designs

The configurations of spent fuel storage pools vary from plant to plant. In BWR's, the pools are located within the reactor building with the bottom of the pool at about the same elevation as the upper portion of the reactor pressure vessel. During refueling the cavity above the top of the pressure vessel is flooded to the same elevation as the storage pool, so that fuel assemblies can be transferred directly from the reactor to the pool via a gate which separates the pool from the cavity. In PWR plants, the storage pool is located in an auxiliary building. In some cases the pool surface is at about grade level, in others the pool bottom is at grade. The refueling cavities are usually connected to the storage pool by a transfer tube. During refueling the spent assembly is removed from the reactor vessel and placed in a container which then turns on its side, moves through transfer tube to storage pool, set upright again and removed from the transfer container to a storage rack. Various gates and weirs separate different sections of the transfer and storage systems. More details concerning various configurations are given in Section 2.3 and Table 1.1.

S.1.5 Selection of Surrogate Cases for More Detailed Studies

Two "older vintage" plants were selected to serve as BWR and PWR surrogates for more detailed studies. The choices, Millstone 1 and Ginna, were based primarily on such factors as availability of data and the relative familiarity of the project staff with the various candidate sites. The

operating histories of the two surrogate plants were modeled to obtain a realistic radioactive inventory in the various spent fuel batches.

S.2 ACCIDENT INITIATING EVENTS AND PROBABILITY ESTIMATES

Accident initiating events that have been considered include

- pool heatup due to loss of cooling water circulation capability,
- structural failure of pool due to seismic events or missiles,
- partial draindown of pool due to pneumatic seal failure, and
- structural failure of pool due to a heavy load drop.

Estimates of the likelihood for each of these initiators are provided in Section 2. It is concluded that the dominant initiators are structural failures resulting from a seismic events ($\sim 2 \times 10^{-5}$/Ry) and heavy load drops ($\sim 3 \times 10^{-5}$/Ry). Uncertainties in the probability estimates are quite large, being at least an order of magnitude in either direction. In the case of seismic events, the seismic hazard and structural fragilities both contribute to the uncertainty range. For heavy load drops, human error probabilities and structural damage potentials are the primary sources of uncertainties.

S.3 EVALUATION OF FUEL CLADDING FAILURE

The SFUEL computer code developed at Sandia National Laboratories (SNL) by Benjamin et al.,[3] analyzes the behavior of spent fuel assemblies after an accident has drained the pool. The analyses predict that self-sustaining oxidation of the Zircaloy cladding (i.e., a cladding fire) would occur for a wide range of decay heat levels and storage geometries. Several limitations in the SFUEL analyses had been recognized in Reference 3 and have been addressed in a modified version of the code, SFUELIW.[4]

The BNL evaluations of SFUELIW have led to the conclusions that the modified code gives a reasonable estimate of the potential for propagation of a cladding fire from high power to low power spent fuel and that the code provides a valuable tool for assessing the likelihood of a catastrophic fire for a variety of spent fuel configurations in the event that the pool is drained.

S.4 CONSEQUENCE EVALUATION

Radioactive releases were estimated for the two surrogate plants for five cladding failure scenarios predicted by SFUEL calculations.

S.4.1 Radioactive Inventories

The radioactive inventories contained in the spent fuel pools (as of April 1987) for Millstone 1 and Ginna were calculated using the ORIGEN2 computer code,[8] based on the operating histories of each of the plants (Appendix A). The calculated data included the 1987 inventories for each fuel batch discharged at each refueling over the operating history.

S.4.2 Release Estimates

Fractional releases for various groups of radionuclides were estimated based on the physical parameters characterizing the SFUEL failure scenario. Thus, four source terms were estimated corresponding to the four accident scenarios.

S.4.3 Off-Site Radiological Consequences

Off-site radiological consequences were calculated using the CRAC2 computer code.[9] Because of several features in the health physics modeling in the CRAC2 code, the population dose results appear to be of limited value. The most meaningful measure of the accident severity appears to be the interdiction area (contaminated land area) which in the worst cases was about two orders of magnitude greater than for core-melt accident. No "prompt fatalities" were predicted and the risk of injury was negligible.

S.5 RISK PROFILE

The likelihood and consequences of various spent fuel pool accidents have been combined to obtain the risks which are summarized in Table S.1. As noted above, the population dose results are of limited value because they are driven by decontamination levels assigned within the CRAC2 code. Thus the

land interdiction area is included in Table 5.1 as a more meaningful representation of severity. The uncertainty in each of these risk indices is estimated to be an order of magnitude in either direction and is due principally to uncertainty in the fragility of the pools and uncertainty in the seismic hazard.

Table S.1 Estimated Risk for the Two Surrogate Spent Fuel Pools from the Two Dominant Contributors

Accident Initiator	Spent Fuel Pool Fire Probability/Ry	Health Risk (Man-rem/Ry)	Interdiction Risk (Sq. Mi./Ry)
Seismic induced PWR pool failure	1.6×10^{-5}	37	8.4×10^{-4}
Seismic induced BWR pool failure	1.8×10^{-6}	4	7.6×10^{-5}
Cask drop* induced PWR pool failure	3.1×10^{-5}	71	.001
Cask drop* induced BWR pool failure	2.5×10^{-6}	6	1.1×10^{-4}

*After removal of accumulated inventory resumes. (Note that many new plants have pool configurations and administrative procedures which would preclude this failure mode.)

The overall risk due to beyond design basis accidents in spent fuel pools for the PWR surrogate plant is about 130 person-rem/Ry and about 12 person-rem/Ry for the BWR surrogate. These estimates are comparable to present estimates for dominant core melt accidents and appear to warrant further attention on this basis alone. However, the unique character of such an accident (substantial releases of long lived isotopes) makes it difficult to compare to reactor core melt accidents. The exposure calculations are driven by assumptions in the CRAC modeling and the results are not sensitive to the severity of the accident. In terms of interdiction area this type of accident has the potential to be much worse than a reactor core melt accident.

Note that the risk results are calculated for two surrogate plants and may not be applicable to generic pool types.

S.6 CONSIDERATION OF MEASURE WHICH MIGHT REDUCE CONSEQUENCES

A number of potential preventive and mitigative measures have been proposed but the only one which is judged to provide a substantial measure of risk reduction is a modification of the spent fuel storage racks themselves. For those plants that use a high density storage rack configuration. Improvement in the air circulation capability is estimated to result in risk reduction up to a factor of ten.

S.7 References for Summary

1. "A Prioritization of Generic Safety Issues," Division of Safety Technology, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, NUREG-0933, December 1983, pp. 3.82-1 through 6.

2. "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, NUREG-75/014 (WASH-1400), October 1975, App. I, Section 5.

3. A.S. Benjamin, D.J. McClosksy, D.A. Powers, and S.A. Dupree, "Spent Fuel Heatup Following Loss of Water During Storage," prepared for the U.S. Nuclear Regulatory Commission by Sandia Laboratories, NUREG/CR-0649 (SAND77-1371), May 1979.

4. N.A. Pisano, F. Best, A.S. Benjamin and K.T. Stalker, "The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool," prepared for the U.S. Nuclear Regulatory Commission by Sandia Laboratories, (Draft Manuscript, January 1984) (Note: the project ran out of funds before the report was published.)

5. IE Bulletin No. 84-03: "Refueling Cavity Water Seal," U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, August 24, 1984.

6. Licensee Event Report, LER No. 84-013-00, Haddam Neck, Docket No. 50-213, "Failure of Refueling Pool Seal," 09/21/84.

7. Nucleonics Week, December 11, 1986, pg. 3-4.

8. A.G. Croff, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Composition and Characteristics of Nuclear Materials," Nuclear Technology, Vol. 62, pp. 335-352, September 1983.

9. L.T. Ritchie, J.D. Johnson and R.M. Blond, Calculations of Reactor Accident Consequences Version 2, CRAC2: Computer Code User's Guide, prepared by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, NUREG/CR-2326 (SAND81-1994), February 1983.

1. INTRODUCTION

Generic Safety Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools," was assigned MEDIUM priority in November 1983.[1] In its prioritization, the NRC staff took account of two factors that had not been considered in earlier risk assessments:[2]

1. Spent fuel is currently being stored rather than shipped for reprocessing or repository disposal, resulting in much larger inventories of spent assemblies in reactor fuel basins than had previously been anticipated; and,

2. A theoretical model[3] suggested the possibility of catastrophic Zircaloy fire, propagating from assembly to assembly in the event of complete drainage of water from the pool.

1.1 Previous Investigations

The Reactor Safety Study[2] (which did not take account of the two factors above) concluded that the risks associated with spent fuel storage were extremely small in comparison with accidents associated with the reactor core. That conclusion was based on design and operational features of the storage pools which made the loss of water inventory highly unlikely, e.g.,

- The pool structures were designed to withstand safe shutdown earthquakes,
- The fuel racks were designed to preclude criticality,
- Pool design and instrumentation precluded inadvertent and undetected loss of water inventory,
- Procedures and interlocks prevented the drop of heavy loads on stored assemblies, and
- The storage structures were designed to accommodate the forces and missiles generated by violent storms.

Probabilities of pool failures due to external events (earthquakes, missiles) or heavy load drops were estimated to be in the range of 10^{-6}/year.

Radioactive release estimates were based on melting of 1/3 of a core for various decay periods, with and without filtration of the building atmosphere (see Ref. 2, Table I 5-2).

Subsequent to the Reactor Safety Study, A.S. Benjamin et al.[3] investigated the heatup of spent fuel following drainage of the pool. A computer code, SFUEL, was developed to analyze thermal-hydraulic phenomena occurring when storage racks and spent assemblies become exposed to air. The computer model takes into account decay time, fuel assembly design, storage racks design, packing density, room ventilation and other variables that affect the heatup of the fuel.

Calculations with SFUEL indicated that, for some storage configurations and decay times, the Zircaloy cladding could reach temperatures at which the exothermic oxidation would become self-sustaining with resultant destruction of the cladding and fission product release. The possibility of propagation to adjacent assemblies (i.e., the cladding would catch fire and burn at a hot enough temperature to heat neighboring fuel assemblies to the ignition point) was also identified. In such cases, the entire inventory of stored fuel could become involved. Cladding fires of this type could occur at temperatures well below the melting point of the UO_2 fuel. The cladding ignition point is about 900°C compared to the fuel melting point of 2880°C.

Uncertainties in the SFUEL calculations were primarily attributed to uncertainties in the zirconium oxidation rates.

Further work was done to refine the SFUEL computer model and to compare calculated results with experimental data.[4] These more recent results have generally confirmed the earlier concepts of a Zircaloy fire which, given the right conditions, will propagate to neighboring assemblies. However, comparisons to out-of-pile heat-up data have not shown good agreement with the code. The authors noted that more work in several areas was needed to define more precisely the conditions and configurations which allow or prevent propagation.

Several studies have been conducted on alternative spent fuel storage concepts. Among these is a report published by the Electric Power Research Institute (EPRI), which applies probabilistic risk assessment techniques to several storage concepts.[5] While this study does not directly address Generic Safety Issue 82; however, it does provide useful insight on appropriate analytical methodology as well as useful data on an in-ground (on-site) storage pool.

1.2 Related Events

There is no case on record of a significant loss of water inventory from a domestic, commercial spent fuel storage pool. However, one recent incident occurred at the Haddam Neck reactor that raised concern about the possibility of a partial draindown of a storage pool as a result of seal failure in the refueling cavity at a time when the transfer tube gates to the pool were open, or when transfer of a spent fuel assembly was in progress.[6]

The Haddam Neck incident occurred during preparations for refueling. An inflatable seal bridging the annulus between the reactor vessel flange and the reactor cavity bearing plate extruded into the gap, allowing 200,000 gallons of borated water to drain out of the refueling cavity into the lower levels of the containment building in about 20 minutes. Gates to the transfer tube and the fuel storage pool were in the closed position, so no water drained from the pool.[7] However, had these gates been open at the time of the leak, and had they not been closed within 10 to 15 minutes, the pool would have drained to a depth of about 8.5 feet, exposing the upper 3 feet of the active fuel region in the spent fuel assemblies.[7] Also, had the transfer of spent fuel been in progress with an assembly on the refueling machine, immediate action would have been necessary to place the assembly in a safe location under water to limit exposure to personnel.

The NRC Office of Inspection and Enforcement required all licensees to promptly evaluate the potential for refueling cavity seal failures.[6] Responses indicated that the refueling cavity configuration at Haddam Neck is unique in that the annulus between the reactor flange and the cavity bearing plate is more than 2 feet wide. In most plants this gap is only 2 inches

wide.[8] About 40 operating (or soon to operate) reactors use inflatable seals. However, because of design differences, the Haddam Neck failure does not appear to be directly applicable to the other plants. It is noted that BWR plants have permanent steel bellows seals to fill the gap between the reactor flange and the cavity bearing plate. This issue is discussed more fully in Section 2.3.

1.3 Risk Potential

The risk potentials of "Beyond Design Basis Accidents in Spent Fuel Pools" are defined in terms of

- the probabilities of various initiating events that might compromise the structural integrity of the pool or its cooling capability,
- the probability of a system failure, given an initiating event,
- fuel failure mechanisms, given a system failure,
- potential radionuclide releases, and
- consequences of a specified release.

This study generally follows the logic of a typical probabilistic risk analysis (PRA); however, because of the relatively limited number of potential accident sequences, the analyses are greatly simplified.

1.4 Discussion of Spent Fuel Storage Pool Designs and Features

The general design criteria for spent fuel storage facilities are stated in Appendix A of 10 CFR 50,[9] and are discussed more fully in Regulatory Guide 1.13.[10]

The pool structures, spent fuel racks and overhead cranes must be designed to Seismic Category I standards. It is required that the systems be designed (1) with capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other

residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.[9]

The configurations of spent fuel storage pools vary from plant to plant. Table 1.1 lists various information about the pools for licensed plants.

In BWRs, the pools are located within the reactor building with the bottom of the pool at about the same elevation as the upper portion of the reactor pressure vessel. (For example, at Oyster Creek the bottom of the pool is at elevation 80'6", and the top at 119'3". The water depth is 38 feet.) During refueling, the cavity above the top of the pressure vessel is flooded to the same elevation as the storage pool, so that fuel assemblies can be transferred directly from the reactor to the pool via a gate which separates the pool from the cavity.

In PWR plants, the storage pool is located in an auxiliary building. In some cases the pool surface is at about grade level, in others the pool bottom is at grade. The refueling cavities are usually connected to the storage pool by a transfer tube. During refueling the spent assembly is removed from the reactor vessel and placed in a container which then turns on its side, moves through transfer tube to storage pool, set upright again and removed from the transfer container to a storage rack. Various gates and weirs separate different sections of the transfer and storage systems. More details concerning various configurations are given in Section 2.3.

1.5 Selection of Surrogate Cases for More Detailed Studies

Two "older vintage" plants were selected to serve as BWR and PWR surrogates for more detailed studies. The choices, Millstone 1 and Ginna, were made somewhat arbitrarily, based primarily on such factors as availability of data and the relative familiarity of the project staff with the various candidate sites. The operating histories of the two surrogate plants were modeled to obtain a realistic radioactive inventory in the various spent fuel batches. Details of the modeling procedures and a listing of the calculated radionuclide content are presented in Appendix A.

It should be noted that both surrogate plants have relatively large inventories of spent fuel assemblies in their spent fuel basins.

1.6 Report Content

Accident initiating events and their probabilities are covered in Section 2. Fuel cladding failure scenarios based on the SFUELIW Computer Code are evaluated in Section 3. Included are sensitivity analyses of the failure scenarios arising from uncertainties in Zircaloy oxidation reaction rate data, and hardware configuration assumptions. Section 4 presents data on the potential for releases of radionuclides under various cladding failure scenarios and compares the projected releases with releases associated with severe core accident sequences. In Section 5, risk profiles are developed in terms of person-rem population doses for several accident sequences. Section 6 considers measures that might mitigate beyond design basis accidents.

1.7 References for Section 1

1. "A Prioritization of Generic Safety Issues," Division of Safety Technology, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, NUREG-0933, December 1983, pp. 3.82-1 through 6.

2. "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, NUREG-75/014 (WASH-1400), October 1975, App. I, Section 5.

3. A.S. Benjamin, D.J. McClosksy, D.A. Powers, and S.A. Dupree, "Spent Fuel Heatup Following Loss of Water During Storage," prepared for the U.S. Nuclear Regulatory Commission by Sandia Laboratories, NUREG/CR-0649 (SAND77-1371), May 1979.

4. N.A. Pisano, F. Best, A.S. Benjamin and K.T. Stalker, "The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool," prepared for the U.S. Nuclear Regulatory Commission by Sandia Laboratories, (Draft Manuscript, January 1984) (Note: the project ran out of funds before the report was published.)

5. D.D. Orvis, C. Johnson, and R. Jones, "Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment," prepared for the Electric Power Research Institute by the NUS Corporation, EPRI NP-3365, February 1984.

6. IE Bulletin No. 84-03: "Refueling Cavity Water Seal," U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, August 24, 1984.

7. Licensee Event Report, LER No. 84-013-00, Haddam Neck, Docket No. 50-213, "Failure of Refueling Pool Seal," 09/21/84.

8. Licensee Responses to NRC IE Bulletin No. 84-03.

9. Code of Federal Regulations, Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities, Appendix A, 'General Design Criteria for Nuclear Power Plants,' General Design Criterion 61, 'Fuel Storage and Handling and Radioactivity Control'."

10. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," December 1981.

Table 1.1 BWR's: DATA ON SPENT FUEL STORAGE BASINS. Included are spent fuel storage inventories as of December 1984, fractions of core in storage, comparisons with the "reference case" of radionuclide inventory, locations of spent fuel basins, and seismic design bases of pools.

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Big Rock Point	240	84	172	2.05	4.9	AB, grd	DBE=0.05g
Browns Ferry-1	3293	764	1068	1.40	46.1	RB, ele	DBE=0.20g
Browns Ferry-2	3293	764	889	1.16	38.2	RB, ele	DBE=0.20g
Browns Ferry-3	3293	764	1768	2.31	76.1	RB, ele	DBE=0.20g
Brunswick-1	2436	560	1056[f]	1.89	46.0	RB, ele	DBE=0.16g
Brunswick-2	2436	560	924[g]	1.65	40.2	RB, ele	DBE=0.16g
Cooper	2381	548	985	1.80	42.9	RB, ele	DBE=0.2g
Dresden-1	700	464	221	0.48	3.36		DBE=0.20g
Dresden-2	2527	724	2014[h]	2.78[h]	70.3[h]	RB, ele	DBE=0.2g
Dresden-3	2527	724	-	-	-	RB, ele	DBE=0.2g
Duane Arnold	1658	368	576	1.57	26.0	RB, ele	DBE=0.12g
Fitzpatrick	2436	560	816	1.46	35.6	RB, ele	DBE=0.15g
Grand Gulf-1	3833	N/A[i]	0	0.00	0.0	N/A	
Hatch-1	2436	560	140	0.25	6.1	RB, ele	DBE=0.15g

Table 1.1 (Cont'd)

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Hatch-2	2436	560	1284	2.29	55.8	RB, ele	DBE=0.15g
Humboldt Bay	220	172	251	1.46	3.2	N/A	DBE=0.50g
LaCrosse	165	72	207	2.88	4.8	AB, grd	DBE=0.12g
LaSalle-1	3323	N/A	0	0.00	0.0	RB, ele	SSE=0.20g
LaSalle-2	3323	N/A	0	0.00	0.0	RB, ele	SSE=0.20g
Limerick-1	3293	N/A	0	0.00	0.0	RB, ele	SSE=0.13g
Millstone-1	2011	580	1346	2.32	46.7	RB, ele	DBE=0.17g
Monticello	1670	484	1137	2.35	39.2	RB, ele	DBE=0.12g
Nine Mile Point-1	1850	532	1244	2.34	43.3	RB, ele	DBE=0.11g.
Oyster Creek	1930	560	1375	2.46	47.5	RB, ele	DBE=0.22g
Peach Bottom-2	3293	764	1361	1.78	58.6	RB, ele	DBE=0.12g
Peach Bottom-3	3293	764	1212	1.59	52.4	RB, ele	DBE=0.12g
Pilgrim-1	1998	580	1128	1.94	38.8	RB, ele	DBE=0.15g
Quad Cities-1	2511	724	1730	2.39	60.0	RB, ele	DBE=0.24g
Quad Cities-2	2511	724	412	0.57	14.3	RB, ele	DBE=0.24g

Table 1.1 (Cont'd)

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Susquehanna-1	3293	764	0	0.00	0.0	RB, ele	SSE=0.1g
Susquehanna-2	3293	764	0	0.00	0.0	RB, ele	SSE=0.1g
Vermont Yankee	1593	368	1174	3.19	50.8	RB, ele	DBE=0.14g
Wash. Nucl.-2	3323	N/A	0	0.00	0.0	N/A	SSE=0.32g

Footnotes

a) Source: U. S. Nuclear Regulatory Commission, Licensed Operating Reactors, NUREG-0020, Vol. 9, No. 1, January 1985.

b) (Stored Assemblies)/(Assemblies in Core).

c) "Reference Source Term" assumes a thermal power of 3000 MWt, stored inventory from ten annual discharges, last discharge six months ago, total inventory 1750 assemblies. Source term relative to "Reference Source Term" has not been corrected for age of fuel in storage.

d) Location: RB = reactor building, AB = auxiliary building, grd = pool at grade level, ele = pool at high elevation in building.

e) Seismic design basis as a function of the gravitational acceleration (g): DBE = design basis earthquake, or equivalent as used for older vintage plants; SSE = safe shutdown earthquake as defined in 10 CFR 100, App. A. Entry shown is the horizontal component.

f) Brunswick-1 has in storage 160 PWR + 656 BWR assemblies, equivalent to 1056 BWR assemblies.

g) Brunswick-2 has in storage 144 PWR + 564 BWR assemblies, equivalent to 924 BWR assemblies.

h) Dresden Units 2 and 3 have two pools in one structure. The data cited are total of the two.

i) N/A = data not available.

Table 1.1 (Cont'd) PWR's: DATA ON SPENT FUEL STORAGE BASINS. Included are spent fuel storage inventories as of December 1984, fractions of core in storage, comparisons with the "reference case" of radionuclide inventory, locations of spent fuel basins, and seismic design bases of pools.

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Arkansas-1	2568	177	388	2.19	56.3	AB, grd	DBE=0.2g
Arkansas-2	2815	177	168	0.95	26.7	AB, grd	DBE=0.2g
Beaver Valley-1	2660	157	104	0.66	17.6	FB, grd	SSE=0.125g
Byron-1	N/A[f]	N/A	0	0.00	0.0	AB, grd	SSE=0.2g
Callaway-1	3411	N/A	N/A	N/A	N/A	AB, grd	SSE=0.2g
Calvert Cliffs-1	2700	217	868[g]	4.00[g]	108.0[g]	AB, grd	DBE=0.15g
Calvert Cliffs-2	2700	217	-	-	-	AB, grd	DBE=0.15g
Catawba-1	N/A	N/A	N/A	N/A	N/A	AB, grd	SSE=0.12g
Cook-1	3250	193	553[g]	2.87[g]	93.1[g]	AB, grd	SSE=0.20g
Cook-2	3411	193	-	-	-	AB, grd	SSE=0.20g
Crystal River-3	2544	177	171	0.97	24.6	AB, grd	SSE=0.10g
Davis Besse-1	2772	177	199	1.12	31.2	AB, grd	DBE=0.15g
Diablo Canyon-1	3338	N/A	N/A	N/A	N/A	AB, grd	DDE=0.4g[j]
Farley-1	2652	157	114	0.73	19.3	AB, grd	SSE=0.10g

Table 1.1 (Cont'd)

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Farley-2	2652	157	62	0.39	10.5	AB, grd	SSE=0.10g
Fort Calhoun	1500	133	305	2.29	34.4	AB, grd	DBE=0.17g
Ginna	1520	121	340	2.81	42.7	AB, grd	DBE=0.20g
Haddam Neck	1825	157	545	3.47	63.4	AB, grd	DBE=0.17g
Indian Point-1	h	0[h]	160	h	h	AB, grd	DBE=0.10g
Indian Point-2	2758	193	332	1.72	47.4	AB, grd	DBE=0.15g
Indian Point-3	3025	193	140	0.73	21.9	AB, grd	DBE=0.15g
Kewaunee	1650	121	268	2.21	36.5	AB, grd	DBE=0.12g
Maine Yankee	2630	217	577	2.66	69.9	AB, grd	DBE=0.10g
McGuire-1	3411	193	91	0.47	16.1	AB, grd	SSE=0.15g
McGuire-2	3411	N/A	N/A	N/A	N/A	AB, grd	SSE=0.15g
Millstone-2	2700	217	376	1.73	46.8	AB, grd	DBE=0.17g
North Anna-1	2775	157	220[g]	1.40[g]	38.9[g]	AB, grd	SSE=0.12g
North Anna-2	2775	157	-	-	-	AB, grd	SSE=0.12g
Oconee-1	2568	177	1037[g]	5.86[g]	150.5[g]	AB, grd	DBE=0.10g

Table 1.1 (Cont'd)

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Oconee-2	2568	177	-	-	-	AB, grd	DBE=0.1g
Oconee-3	2568	177	218	1.23	31.6	AB, grd	DBE=0.1g
Palisades	2530	204	480	2.35	59.5	AB, grd	DBE=0.20g
Palo Verde-1	N/A	N/A	N/A	N/A	N/A	AB, grd	SSE=0.20g
Point Beach-1	1518	121	524[g]	4.33[g]	65.7[g]	AB, grd	DBE=0.18g
Point Beach-2	1518	121	-	-	-	AB, grd	DBE=0.18g
Prairie Island-1	1650	121	601[g]	4.97[g]	82.0[g]	AB, grd	SSE=0.12g
Prairie Island-2	1650	121	-	-	-	AB, grd	SSE=0.12g
Rancho Seco-1	2772	177	260	1.47	40.7	AB, grd	SSE=0.25g
Robinson-2	2300	157	152	0.97	22.3	AB, grd	DBE=0.20g
Salem-1	3338	193	296	1.53	51.2	AB, grd	DBE=0.20g
Salem-2	3411	193	265	1.37	46.8	AB, grd	DBE=0.20g
San Onofre-1	1347	157	94	0.60	8.1	AB, grd	DBE=0.50g
San Onofre-2	3410	217	217	1.00	34.1	AB, grd	SSE=0.67g
San Onofre-3	3390	217	0	0.00	0.0	AB, grd	SSE=0.67g

Table 1.1 (Cont'd)

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[c]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Sequoyah-1	3411	193	65	0.34	11.5	AB, grd	SSE=0.18g
Sequoyah-2	3411	193	130	0.67	23.0	AB, grd	SSE=0.18g
St. Lucie-1	2700	217	352	1.62	43.8	AB, grd	DBE=0.10g
St. Lucie-2	2560	N/A	N/A	N/A	N/A	AB, grd	SSE=0.10g
Summer-1	2775	157	52	0.33	9.2	AB, grd	SSE=0.15g
Surry-1	2441	157	608[g]	3.87[g]	94.5[g]	AB, grd	SSE=0.15g
Surry-2	2441	157	-	-	-	AB, grd	SSE=0.15g
Three Mile Island-1	2535	177	208	1.18	29.8	AB, grd	DBE=0.12g
Three Mile Island-2	f	177	0	0.00	0.0	AB, grd	SSE=0.12g
Trojan	3411	193	312	1.62	55.1	AB, grd	DBE=0.25g
Turkey Point-3	2200	157	445	2.83	62.4	AB, grd	DBE=0.15g
Turkey Point-4	2200	157	430	2.74	60.3	AB, grd	DBE=0.15g
Waterford-3	N/A	N/A	N/A	N/A	N/A	AB, grd	SSE=0.10g
Yankee Rowe	600	76	250	3.29	19.7	AB, grd	None

Table 1.1 (Cont'd)

Plant	Thermal Power (MWt)	Number of Fuel Assemblies in Core[a]	Spent Fuel Stored Inventory[a] (No. of Assemblies)	Stored Inventory Fractions of Core[b]	Radioactivity Relative to Reference Case[c] (per cent)	Storage Pool Location[d]	Seismic Design Basis[e]
Zion-1	3250	193	863[g]	4.47[g]	145.3[g]	AB, grd	SSE=0.17g
Zion-2	3250	193	-	-	-	AB, grd	SSE=0.17g

Footnotes for Table 2

a) Source: U. S. Nuclear Regulatory Commission, Licensed Operating Reactors, NUREG-0020, Vol. 9, No. 1, January 1985.

b) (Stored Assemblies)/(Assemblies in Core).

c) "Reference Source Term" assumes a thermal power of 3000 MWt, stored inventory from ten annual discharges, last discharge six months ago, total inventory 700 assemblies. Source term relative to "Reference Source Term" has not been corrected for age of fuel in storage.

d) Location: RB = reactor building, AB = auxiliary building, FB = fuel building, g = pool at grade level, e = pool at high elevation in building.

e) Seismic design basis as a fraction of the gravitational acceleration (g): DBE = design basis earthquake, or equivalent as used for older vintage plants; SSE = safe shutdown earthquake as defined in 10 CFR 100, App. A. Entry shown is the horizontal component.

f) N/A = data not available.

g) Spent fuel basin shared by two units. Entries shown are totals.

h) Indian Point-1 is permanently shutdown.

i) TMI-2 is indefinitely shutdown.

j) Diablo Canyon originally used the "Double Design Earthquake," DDE acceleration = 2 DBE. Later, more elaborate analysis was done to postulate an earthquake of 0.5g associated with the Hosgri Fault.

2. ACCIDENT INITIATING EVENTS AND PROBABILITY ESTIMATES

2.1 Loss of Water Circulating Capability

The spent fuel basins of U.S. nuclear power stations contain a large inventory of water, primarily to provide ample radiation shielding over the top of the stored spent fuel. Some typical pool dimensions and water inventories are shown in Table 2.1. The heat load from decay heat of spent fuel depends on decay time since the last refueling. Heat loads for the entire spent fuel inventory of the two older vintage surrogate plants are shown in Table 2.2 (data extrapolated to the 1987 scheduled refuelings). The cooling systems provided for spent fuel pools typically have a capacity in the range of 15 to 20×10^6 Btu/hr (4.4 to 5.9×10^3 kw).

In the event that normal circulation of the cooling water is disrupted, e.g., due to station blackout, pump failure, pipe rupture, etc., the water temperature of the pool would steadily increase until bulk boiling occurred. (Note: In a situation where the stored inventory was small, an equilibrium temperature, below the boiling point, would be reached at which surface evaporation balanced the decay heat load).

Thermal-hydraulic analyses of the consequences of partial or complete loss of pool cooling capability are a routine part of the safety analysis reports required for licensing and amendments thereto. Generally, these analyses consider several scenarios ranging from typical to extremely conservative conditions. A sampling of conservative results for several plants is given in Table 2.3. The data clearly demonstrate that the time interval from loss of circulation until exposure of fuel to air is quite long. Even in the most pessimistic case cited in Table 2.3 (Docket No. 50-247), the water level in the pool would drop only about 6 inches per hour. Thus, there appears to be considerable time available to restore normal cooling or to implement one of several alternative backup options for cooling.

For licensing purposes, it has been accepted that the time interval for restoring cooling manually from available water sources is adequate without requiring active (automatic) redundant cooling systems.

However, in considering the prioritization of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools," the NRC staff recognized that there is a finite probability that cooling could not be restored in a timely manner.[2] The case treated in Ref. 2 was for a BWR. The estimated frequency for the loss of one (of two) cooling "trains" was taken to be 0.1/Ry (the value assumed in WASH-1400).[3] This combined with the conditional probabilities of failure/non-availability of the second "train" yielded a combined frequency of a pool heatup event of 3.7×10^{-2}/Ry. (This estimate appears to be somewhat conservative since no "pool heatup events" are on record after $\sim 10^3$ reactor years of accumulated experience).

To escalate from a "pool heatup event" to an event which results in fuel damage requires the failure of several alternative systems that are capable of supplying makeup water (the RHR and condensate transfer systems, or, as a last resort, a fire hose). Estimated frequencies of failure for each of the alternatives, combined with the frequency of a pool heatup event, resulted in an estimated frequency of 1.4×10^{-6}/Ry for an accident initiated by loss of spent fuel pool cooling.

Originally, the spent fuel pool at the Ginna plant had only one installed cooling train with a "skid-mounted" backup pump and heat exchanger. However, a second cooling train was to have been installed in 1986.[4] Because of the third option for cooling at Ginna (the skid-mounted system) the probability estimate for an accident initiated by a pool heatup event should be reduced to 5×10^{-7}/Ry, i.e., about a factor of 3 smaller than for the BWR case analyzed in Ref. 2.

2.2 Structural Failure of Pool

Because of the massive reinforced concrete structure of LWR spent fuel storage pools, designed to Category I seismic criteria, initiating events that would lead to a structural failure are extremely unlikely. On the other hand, a structural failure that resulted in rapid and complete draining of water from the pool would have serious consequences. Probabilities of events that might result in loss of structural integrity are estimated in the following two subsections.

2.2.1 Structural Failure of Pool Resulting from Seismic Events

Procedures and conventions for a detailed probabilistic risk assessment (PRA) of seismically-induced core damage accident sequences have been presented in Ref. 5. The recommended methodology could be applied to spent fuel pools as a separate plant component, or could be coupled to a core damage sequence that might occur simultaneously during a severe earthquake. To date, the seismic PRA methodology has not been rigorously applied to spent fuel pools.

Seismic risk analyses consist of three basic steps:

1) portrayal of the seismic hazard in terms of annual frequency of exceedance as a function of some ground motion parameter (e.g., the peak ground acceleration);

2) assessment of the probability that the capacity of a structure or component can survive the seismic event, often expressed in the form of a fragility curve which is the inverse of the capacity for survival; and, finally,

3) a logic model, e.g., an event tree, which relates a seismic-induced failure to a higher order event that results in some category of radioactive release.

In principle, an appropriate convolution of the probability functions derived in steps 1) and 2) yields a probability function for seismic-induced failure. It is recognized that large uncertainties exist in the two input probability functions which are reflected in the function expressing the probability of failure.

The three steps and the treatment of the uncertainties have been summarized by Reed,[6] who notes that the largest uncertainties are associated with step 1), i.e., the probabilities of occurrence of severe earthquakes having correspondingly very large ground accelerations. Reed makes the assertion that "due to the large uncertainties in the ground shaking hazard, it is

unproductive to refine the structure and equipment capacity calculations to accuracies which are inconsistent with the hazard uncertainty."[6] The specific applicability to spent fuel pools of Reed's assertion is discussed in Section 2.2.1.3.

2.2.1.1 A Review of Seismic Hazard Data

The primary difficulty in characterizing the seismic hazard at specific sites in the Eastern United States (EUS), i.e., sites to the east of the Rocky Mountains is that severe earthquakes are rare events in the EUS. A systematic analysis of recorded earthquakes and their relationship to geological features has yielded seismic zonation maps of the EUS.[7] However, such information cannot readily be translated into the type of seismic hazard functions needed as input for PRA. Consequently, available historical data are insufficient for obtaining meaningful site specific estimates of the frequency of severe events.

A different approach to seismic hazard analysis has been developed at Lawrence Livermore National Laboratory (LLNL) by D.L. Bernreuter and his colleagues under NRC sponsorship. The initial study was a part of the NRC's Systematic Evaluation Program (SEP).[8] The methodology has been further refined in a subsequent study, "EUS Seismic Hazard Characterization Project" (SHC).[9,10]

Since the SEP and SHC results will be used for the seismic hazard estimates, some further discussion of the Bernreuter methodology is appropriate. Three basic steps are involved:

1. the elicitation of expert opinion to delineate and characterize seismically active zones in the EUS;

2. using the input data of each expert, the computation of the seismic hazard functions at specific reactor sites using several alternative ground motion attenuation models with site corrections, and integrating over each of the delineated seismic zones; and, finally,

3. the combining of the separate expert data accompanied by the generation of uncertainty limits from the spread in expert opinions and from the self-evaluations of each expert on the degree of confidence in the input opinion.

The various steps are carried out in a highly disciplined and systematic manner. Provision is made at various stages for peer review of the methods and input opinion, feedback to the experts and critical evaluation of the results.

In step 1, each expert prepares a "best estimate" map which delineates the seismic zones. Each zone is characterized by a set of parameters that give the maximum earthquake intensity to be expected for that zone (upper magnitude cut-off), the expected frequency of earthquakes, and the magnitude recurrence relation. For each input (zone boundaries, seismic parameters), the expert provides a measure of his degree of confidence. Also each expert is given the option of submitting alternative maps of differing zonations and characterizations (up to as many as 30 maps). The data from each expert are evaluated separately through step 2.

In step 2, the contribution at a given site from each zone is integrated over the zone area and then over all zones. This requires the use of ground motion models for which a range of alternative models are employed to yield a set of alternative hazard curves. A "Ground Motion Panel" of experts have selected several alternative models to be used, each having a weighting factor (see Ref. 9, App. C). Also each ground motion model incorporates a site specific correction to account for local geology.

In step 3, the results of the individual experts are combined to obtain a "best estimate" hazard curve and the uncertainty bands are computed in several alternative ways.

It is obvious that the methodology requires a massive data collection and computer effort. In its present state, the final results are not in a form to be easily applied to a specific PRA by a non-expert in seismology. Further work is needed to develop a more convenient format for presenting the final

results.[11] In particular, numerical tabulations of the sets of hazard curves (such as those shown in Figs. 2.1 and 2.2) and their derivatives, $|dH/da|_i$ for each reactor site would be helpful. Also, it appears that the local site geology needs more rigorous consideration in the derivation of the hazard curves (see below).

Members of the Peer Review Panel have suggested several ways in which the methodology could be refined (see Ref. 10, Section 7 and Appendices D.1-D.4). Many of these suggestions were implemented in the final feedback process and were included in the final results reported in Ref. 10. In general, the reviewers agreed that the results are "credible and as good as present scientific understanding of eastern U.S. (EUS) seismicity probably allows" (Ref. 10, App. D.1).

Comments from NRC licensees and their consultants indicated objections to application of the results to specific sites, noting that the site specific correction factors in the ground motion models were too simplified to adequately take local geological factors into account (Ref. 10, App. D.6). Also, the criticism was made that the results were not adequately tested against recent historical records.

In order to illustrate the hazard curves, their range of uncertainties and comparison with other studies, a series of figures taken from Ref. 10 for the Millstone site is reproduced in Figs. 2.1-2.4.

Figure 2.1 is the hazard curve obtained from combining the "best estimate" results for all experts in the SHC study (including the seismic and the ground motions panels). The curve plots frequency of exceedance per year vs. peak ground acceleration.

Figure 2.2 illustrates the uncertainties in the hazard curve (15, 50, and 85 percentiles) derived from the spread in expert opinion and the self-confidence factors in the input parameters. It can be seen that the spread between the 15 and 85 percentiles is about a factor of 20 at low PGA increasing to about 350 at the high PGA. Comparison of Figs. 2.1 and 2.2 shows that

the "best estimate" curve is considerably higher than the 50 percentile, i.e., the mean > median.

Figure 2.3 illustrates the spread in the "best estimate" hazard curves for all of the experts participating in either the SEP[8] or the SHC[9] studies, or both (6 experts participated in both studies). The spread ranges from about one order of magnitude at lower PGA to about 1.5 orders of magnitude at the higher PGA. The curve marked "A," which falls considerably below the main grouping, was derived from data input in the SEP study by one of the experts who participated in both studies. This revised input for the SHC project raised the derived curve by an order of magnitude at the low accelerations and by about two orders of magnitude at the higher PGA, this raises the obvious question of whether the experts were somehow influenced by the opinions of their colleagues, or whether the revision resulted from a more careful consideration of the various geological factors that were taken into account in preparing the input parameters. The question of testing the results for inadvertent biases of this nature was addressed by the Peer Review Panel members, but their recommendations could not be fully implemented in the final report due to limited time and budget (Ref. 10, pg. 7-3).

Figure 2.4 compares the "best estimate" hazard curves for the individual SHC experts with curves generated from zonation maps prepared by the U.S. Geological Survey (USGS)[12] and historical data of the past 280 years. As can be seen, the USGS hazard curve (denoted by "X") lies above the SHG data. Bernreuter et al. attribute the difference between the SHC and the USGS curves to the variations in the equations used for conversions from intensity to magnitude and in the values for the rate of earthquake recurrence (Ref. 10, pg. 8-1 et seq.). As would be expected the 280 year historical hazard curve (denoted by "H") falls below the SHC data because it does not include postulated stronger earthquakes with return times much greater than the time span of the historical record.

It should be noted that recent research has raised significant questions concerning the frequency of strong earthquakes in the coastal zone of the EUS.[13] The speculation has arisen from paleoseismic field studies originally focused on the region of the strong earthquake near Charleston, SC, in 1886,

which produced many "sand blows."[14,15] These result from the liquefaction and venting to the surface of sub-surface water-saturated sediment. Several sand blow craters have been found for which radiocarbon dating indicates that moderate to large earthquakes have recurred in the Charleston region on an average of about every 1800 years.[16] The latest (prior to 1886) occurred about 1100 years ago.[16] Sand blows from prehistoric earthquakes have been unearthed recently in the region extending from near Savannah, GA as far north as Myrtle Beach, SC.[17] The broad extent of sand blows suggests that Charleston-type earthquakes might be associated with some tectonic feature which extends for some distance along the east coast and not uniquely centered near Charleston. Up to the present time, no systematic field search has been made for sand blows outside of the Savannah to Myrtle Beach region.[18] Recently Thorson et al. reported the existence of apparent sand blow craters in eastern Connecticut.[19] These craters were recently examined by a USGS field team and assessed as not being of the same nature as those observed in South Carolina.[18]

2.2.1.2 Seismic Hazard Estimates for the Millstone and Ginna Sites

The "best estimate" and the median, 15 and 85 percentile seismic hazard curves developed by the SHC project for the Millstone site are shown in Figs. 2.1 and 2.2.[10] These four hazard curves were used to develop the estimates of the seismic failure probabilities of Millstone 1, as described in Section 2.2.1.4 below.

Hazard curves, such as shown in Figures 2.1 and 2.2, are expected to have some upper limit cutoff, i.e., PGA's which would never be exceeded. We have assumed that the upper limit cutoff for the Millstone site occurs at approximately 1 g (980.7 cm/sec^2), but a different cutoff would give a substantially different pool failure frequency.

Seismic hazard curves for the Ginna site were not generated in the SHC project;[20] however, the SEP project included data for Ginna.[8] Unfortunately, the format of the SEP results, which were directed primarily at obtaining site specific spectra, cannot readily be translated into a "best estimate" hazard curve. In want of a better procedure, we have synthesized a hazard curve for

Ginna from the Millstone curve, using ratios of PGA's for 200, 1000, and 4000 year return times, tabulated for the two plants in the SEP study.[8] The hazard curve resulting from this synthesis is shown in Fig. 2.5. Because of the higher upper magnitude cutoff at the Ginna site, as perceived by experts, (Millstone: MMI = 8.0 vs. Ginna: MMI = 8.2), we have assumed the upper cutoff PGA of the hazard curve to be 1.25g. Although this is recognized to be a somewhat pessimistic assumption, it serves the useful purpose of illustrating the sensitivity of the calculated seismic risk to the upper cutoff of the hazard curve.

2.2.1.3 Seismic Fragility of Pool Structures

Fragility curves specifically for spent fuel pools have never been developed.[21] It is necessary therefore, to rely on fragility assessments for other structures which appear to be of similar construction to spent fuel storage pools. It must be recognized that this procedure introduces an additional element of uncertainty in the final risk estimates -- an uncertainty that is difficult to quantify. Another source of uncertainty is the degree to which the stainless steel lining of a pool would enhance the seismic strength capacity (i.e., reduce the fragility). Conceivably, the reinforced concrete structure of the pool could crack without loss of integrity of the pool lining.

The dilemma of selecting an appropriate fragility for a BWR plant is aggravated by the fact that the pool structure extends typically from the 60 to the 100 foot elevations above grade with the resultant amplification of the seismic bending stresses relative to the lower elevations of the structure.[22]

For the present analyses, two, somewhat diverse sets of fragility estimates, have been used:

1) the fragility curve developed by R.P. Kennedy et al.[23] for the Oyster Creek reactor building; and

2) the fragility of the Zion plant auxiliary building shear walls (north-south ground motion).[24] .

In each case, the fragility curve is defined by the following equation:

$$F(a) = \Phi\,[(\ell n\ a/\breve{A}/\beta_R]\ , \qquad (2.1)$$

where F(a) is the probability of structural failure given a peak ground acceleration, PGA = a. $\Phi(\cdot)$ is the normal distribution function, $\breve{A}$ is the median fragility level (i.e., the acceleration at which there is a 50% probability of failure) and β_R is the logarithmic standard deviation expressing the randomness in the value of $\breve{A}$. A third parameter, β_U, is used to express the uncertainty in the median value and is used to generate upper and lower confidence limits. For example, it can be shown that the substitution for $\breve{A}$ in Eq. 2.1 of $\breve{A}' = \breve{A}\ e^{+\beta_U}$ and $\breve{A}' = e^{-\beta_U}$ generate respectively the 84 and 16 percentile curves.

Thus, a set of fragility curves can be generated from three parameters, $\breve{A}$, β_R and β_U. The data used for generating the "Kennedy" and the "Zion" curves are given in Table 2.4.

Kennedy notes that the estimated median fragility value of about 0.75 g is considered applicable to plants designed in the U.S. in the mid 1960's. The Kennedy fragility curve is shown in Fig. 2.6, with the 84 and 16 percentile limits. The corresponding Zion curves appear in Fig. 22, pp. 3-35 of Ref. 24.

2.2.1.4 Seismically-Induced Failure Probabilities

The convolution of the derivative of a seismic hazard curve (e.g., Fig. 2.1) with a fragility curve, yields the annual probability of a seismically-induced failure. This can be expressed by the equation:

$$P_{i,j} = \int_0^{a_{max}} \left|\frac{dH}{da}\right|_i F(a)_j\ da\ , \qquad (2.2)$$

where $P_{i,j}$ is the failure probability obtained from the convolution of hazard curve i with fragility curve j, $|dH/da|_i$ is the derivative of the hazard curve i (i.e., the annual frequency of occurrence of peak ground acceleration, a, and $F(a)_j$ is failure probability at acceleration, a, for

fragility curve j. The integration is cut off at the upper limit expected for the PGA. Since the seismic hazard curve is not an analytic function, the derivative dH/da and the integration are carried out numerically.

Given many hazard and fragility curves from which to choose, and there being no a priori basis for choosing a particular pair, the convolution expressed in Eq. 2.2 can be carried out for each pair of curves with weighting factors assigned to each of the curves in each set. The resultant collection of $P_{i,j}$ gives a probability distribution which expresses the uncertainties in the analysis. The probability density distribution obtained for the Millstone site is shown in Fig. 2.7.

At least in principle, the various hazard and fragility curves (sets i and j) do not have an equal likelihood of being correct. Therefore, a weighting factor (ω_i or ω_j) should be assigned to each curve which reflects an "engineering judgement" of its relative validity. The mean probability for failure is then derived from the following expression,

$$\overline{P}_f = \sum \omega_i \omega_j P_{i,j} / \sum \omega_{i,j} , \qquad (2.3)$$

where $\sum \omega_i = 1$, $\sum \omega_j = 1$ and $\sum \omega_{i,j} = \sum \omega_i \omega_j = 1$. The weighting factors assigned by BNL for the Millstone case are given in Table 2.5. As can be seen from the table, the "best estimate" hazard curve has been assigned a weighting factor of 0.5 with the remaining 0.5 distributed among the median, 15 and 85 percentile curves. The "Kennedy" set of fragility curves were assigned a total weighting factor of 0.75 with the remaining 0.25 distributed among the "Zion" set. Assuming an upper limit cutoff of 1.0g, the mean probability of failure, $\overline{P}_f$, derived from the 24 sets of $P_{i,j}$, using the weighting factors listed in Table 2.5 and Equation 2.3, was

$$\overline{P}_f = 2.2 \times 10^{-5}/\text{year (Millstone)} .$$

In the case of Ginna, only a single hazard curve (Fig. 2.5) was used, there being insufficient data to generate median, 15 and 85 percentile curves for this site. Because of the structure of the Ginna spent fuel pool, the "Zion" fragility curves are more appropriate, than the "Kennedy" curves.

Therefore, higher weighting factors were assigned to the "Zion" curves as shown in Table 2.5. Based on an upper limit PGA cut-off of 1.25g, the mean probability resulting from the convolution of the single hazard curve with the six weighted fragility curves was

$$\bar{P}_f = 1.6 \times 10^{-5}/\text{year (Ginna)} .$$

The difference between the estimates for Millstone and Ginna, 2.2×10^{-5} vs. 1.6×10^{-5}, should not be regarded as highly significant, but more as an indication of the sensitivity of the results to the weighting factors assigned to the fragility curves.

2.2.2 Structural Failures of Pool Due to Missiles

Missiles generated by tornadoes, aircraft crashes or turbine failure could penetrate the pool structure and result in structural failure.

The probability of tornado missiles depends on the frequency of tornadoes at the site, the target area presented to the missile and the angle of impact. An analysis made by Orvis et al.[25] for an average U.S. site derives a probability of $<1 \times 10^{-8}$/year for structural loss of pool integrity due to a tornado missile (Ref. 25, pg. 4-44).

Similarly, the analysis for structural failure of a pool from an aircraft crash yielded a probability of $<1 \times 10^{-10}$/year (Ref. 25, pg. 4-58).

The damage caused by Missiles generated by turbine failure depends on the orientation of the turbine axis relative to the structure, as well as the frequency of turbine failure. An analysis by Bush yields a probability of $\sim 4 \times 10^{-7}$/year for spent fuel pool damage from a turbine failure missile.[26] In the case of Ginna, the probability would be several orders of magnitude smaller (i.e., essentially zero) because the spent fuel pool is shielded from turbine missiles by the primary containment.

2.3 Partial Draindown of Pool Due to Refueling Cavity Seal Failures

On August 21, 1984, the Haddam Neck Plant experienced a failure of the refueling cavity water seal, while preparing for refueling. The water level

in the refueling cavity dropped by about 23 feet to the top of the reactor vessel flange within 20 minutes -- a loss of approximately 200,000 gallons, or a leak rate of about 10,000 gallons per minute.[27] At the time of the event, refueling had not begun. The gates of the transfer tube connecting the refueling cavity to the spent fuel storage pool were closed.

Although the seal failure did not result in an accident or in the release of radioactivity, the incident raised the question of whether similar failures might occur while spent fuel was being transferred or while transfer gates to the spent fuel basin were open, either case of which might result in exposure of spent fuel to air and possible clad failure.

All licensed plants were instructed to evaluate the potential for and consequences of a refueling cavity seal failure.[27]

Refueling cavity seals, seal the gap between the reactor vessel flange and a flange on the inner periphery of the reactor cavity, or the floor of the cavity.

BWR's have a permanently installed stainless steel bellows to seal the gap, and are, thus, not subject to failure of the Haddam Neck type.

Many PWR's seal the gap with gaskets held down by a bolted flat steel ring. Such systems have experienced difficulties in achieving tight seal because of surface irregularities and small vertical and concentric offsets in the two flanges. Consequently, many plants have converted to inflatable (pneumatic) rubber seals. Also, it should be noted that pneumatic rubber seals are often used to seal the gates in transfer tubes or canals.

Licensee responses to the IE Bulletin indicate that the Haddam Neck cavity configuration is unique in that the width of the annular gap between the reactor flange and the cavity flange is about two feet, whereas, in most plants the gap is of the order of <1" to -3". As of summer 1985 some 45 units used pneumatic seals in the refueling cavity.[28]

Typical pneumatic seals are illustrated in Figures 2.8-2.10. There are many variations in the details of the designs, e.g., some plants have various types of retainers to support the rubber seals (e.g., see Figure 2.10), others rely on the rubber seal alone (e.g., see Figure 2.9). According to the responses of the licensees, even if a pneumatic seal should deflate, the leakage would be expected to be small or negligible, because the wedged shaped upper section would maintain a good seal (refer to Figure 2.8), i.e., the deflated seal would not distort enough under the hydrostatic head to extrude through the gap.

Aside from the Haddam Neck 1984 incident, a few cases have been reported in which inflated seals have failed, either in the refueling cavity or transfer gates. None of these events had significant radiological consequences. Several such events are listed in Table 2.6. It is likely that this list is not exhaustive. To the best of the authors' knowledge no data base has been compiled (or is available) of the failure rate of pneumatic seals and their pressurizing systems of the types used in nuclear power plants, or of similar seals used in non-nuclear industries.

Based on the limited experience cited in Table 2.6, the historical failure rate in seals/systems is in the range of $\sim 1 \times 10^{-2}$/Ry. Because of advances in design, increased awareness and surveillance, the present failure rate is estimated to be an order of magnitude smaller, i.e., $\sim 1 \times 10^{-3}$/Ry.

As is obvious from Table 2.6, a seal failure does not necessarily result in the rapid loss of water inventory from spent fuel transit or storage locations. The limited experience indicates that the most probable time for a refueling cavity seal to fail is shortly after installation, while the cavity volume is being filled with water. According to the analyses supplied by licensees in response to IE Bulletin No. 84-03, the failure of a pneumatic refueling cavity seal in most PWR plants would not result in massive leaks because of the relatively narrow gap to be sealed and the geometric shape of the seal. Also, leaks from seal failures in transfer tube/canal gates would be limited, in most cases, because the leakage would be into a confined volume, e.g., from the storage pool into a drained up-ender sump. Taking these factors into consideration, it is estimated that the frequency of a

serious loss of pool water inventory resulting from a pneumatic seal failure to be in the range of ~1×10^{-5}/Ry.

Even a large loss of water inventory from the spent fuel pool does not necessarily result in uncovering and subsequent failure of fuel. Most spent fuel basins are constructed with weirs below the transfer gates which preclude complete drainage of the pool, even in the event of a catastrophic Haddam Neck type failure with the transfer tube/canal gates open. In most cases, the water level would remain a foot or more above the active zone of the spent fuel assemblies. In a few cases, the upper several inches of the fuel could uncover. (Note: Licensee responses to IE Bulletin 84-03 did not always provide information about the elevations of weirs and tops of stored assemblies.)

In the event of a draindown of the pool to near the top of the fuel assemblies, there would still be time (1/2 to 1 hour) to close gates and restore a supply of water before the residual water inventory reached the boiling point. However, as noted in one licensee response, even if the fuel remained covered "dose rate in the vicinity of the spent fuel pool would, however, be high, complicating recovery from the event."[29]

A pool heatup event similar to the partial draindown scenario described above was considered by the NRC staff in Ref. 2. A conditional probability for failure to restore adequate makeup water was taken to be 5×10^{-2}, based purely on judgement. Because of higher radiation levels in the partial draindown scenario, it is estimated that the probability of failure to restore adequate makeup water to be somewhat larger, i.e., ~1×10^{-1}.

Given all of the above, the probability of a pneumatic seal failure which results in exposure to air of stored spent fuel with resulting clad failure is estimated to be of the order of

$$P = 1 \times 10^{-6}/\text{Ry}.$$

2.4 Pool Structural Failure Due to Heavy Load Drop

WASH-1400 considered the probability of structural damage to the pool due to the dropping of a fuel transfer cask (Ref. 3, pg. I-97). In the analysis, it was anticipated that one spent fuel shipment per week would be the equilibrium shipping rate. The estimated rate for a drop resulting in pool failure (for a single unit plant) was 4.5×10^{-7}/Ry.

The above frequency was based on a crane failure probability of 3×10^{-6} per operating hour. It was further assumed that each lift was of 10 minutes duration and for a 10 second period per lift the cask would be in a position to cause gross structural damage to the pool wall if a crane failure occurred. Human error was not considered.

Since spent fuel is not currently being shipped, this hazard does not exist at the present time. However, at some point in the future, spent fuel will have to be removed from the reactor pools, either to some onsite storage facility, or eventually to a high level waste repository. At that time, the frequency of removal of spent fuel will be correspondingly greater.

Orvis et al.[25] have reexamined the cask drop probability and have used the following probabilities:

Mechanical failure of crane = 3×10^{-6}/operating hour --
Electrical control failure of crane = 3×10^{-6}/operating hour
Human error = 6×10^{-4}/lift.

As can be seen, human error dominates the Orvis estimates for probability of a cask drop. The Orvis datum for human error was based on a study by Garrick et al.[30] which concerned human reliability in the positioning of heavy objects. The applicability of the Garrick study to crane operations is not obvious. Nevertheless, a human failure rate in the range of 10^{-3} to 10^{-4} per operation appears to be consistent with data listed in the NRC handbook on human reliability analysis[31] for cases in which the operation has one or more people who serve as "checkers" and involves some degree of personal risk to the operating personnel.

Obviously, not all human failures associated with the lifting and moving of a spent fuel shipping cask would result in structural damage to the pool. The section of the pool where the cask is set down has an impact pad to absorb the impulse of a dropped cask. Accidents in unloading the cask from or reloading on the transport vehicle would not involve the pool.

Only horizontal movements of the cask above a structurally critical section of the pool would pose the threat of structural damage. As noted above, WASH-1400 assumed that the sensitive section is the vertical wall at the pool edge. It was implicitly assumed that all load drops on the pool edge would result in structural failure. This assumption appears to be too simplistic and consequently too conservative for the following reasons:

- many "load drops" would be partially attenuated by crane mechanisms which limit descent rates, and reduce impact energy,
- in case of some "off-center" hits, the full potential impact energy would not be absorbed by the pool edge (cask tilted, one end strikes floor first), and
- account should be taken of exterior cask fittings (e.g., cooling vanes) which absorb some impact energy.

No rigorous structural analyses have been performed to scope the range of damage to a pool edge from a cask drop. In the absence of such analyses, it has been necessary to estimate the conditional probability of catastrophic structural damage given a cask drop in the vicinity of the pool edge. It is estimated that the conditional probability is less than 100% and greater than 1%. A conditional probability of 10% has been arbitrarily selected for the hazard calculation and 100% and 1% used for defining the range of uncertainties.

Since human error, rather than mechanical or electrical failure, appears to dominate the hazard arising from shipping cask movements, the various steps in the crane operation have been identified in Table 2.7, which also lists the types of human error associated with each step. The distribution of failure

frequency in the various steps has been estimated and listed in the last column of Table 2.7. (This distribution was subjected to "peer review" by BNL rigging personnel and managers who oversee operations of this type.)

It will be noted that most steps in the crane operation do not jeopardize the structural integrity of the pool. Only in steps 5a and 5b (see Table 2.7) could the cask strike the pool edge. An accident of the type listed in 5a (horizontal movement with cask not high enough to clear the pool edge) would probably not cause serious damage because of the limited kinetic energy of the cask associated with the slow velocity of horizontal crane movements. Thus, only step 5b in Table 2.7 is considered in the hazard calculation.

For purposes of calculating the cask drop hazard, i.e., the probability of catastrophic structural damage to the pool resulting from a cask dropping on the pool edge, the assumptions listed in Table 2.8 were used. Table 2.8 also lists the uncertainty ranges for each of the parameters. The results are as follows:

Probability of structural failure due to cask drop on pool edge caused by mechanical or electrical failure of crane = 3.5×10^{-7}/Ry.

Probability of structural failure due to cask drop on pool edge caused by human error = 3.1×10^{-5}/Ry.

If the failure rates summarized in Table 2.8 are assumed to be statistically independent, then the uncertainty in the overall failure rate is dominated by the uncertainty in the probability of pool failure. Thus the overall uncertainty is about a factor of ten in either direction.

2.5 Summary of Accident Probabilities

The probability estimates made in Sections 2.1-2.4 are summarized in Table 2.9. These include only those accidents that result in the complete loss of pool water inventory. It will be seen that shipping cask drop resulting from human error and seismic induced failures dominate in the hazards. As

previously discussed the uncertainty in both of these probabilities is quite large and has been estimated to be an order of magnitude in either direction.

2.6 References for Section 2

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4. Rochester Gas and Electric Corporation, Docket No. 50-244, "Design Criteria, Ginna Station, Spent Fuel Cooling System," EWR 1594, Revision 1, October 10, 1979; U.S. Nuclear Regulatory Commission, Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 65 to Provisional Operating License No. DPR-18, Rochester Gas and Electric Corporation, R.E. Ginna Nuclear Plant, Docket No. 50-244, November 14, 1984.

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10. D.L. Bernreuter, J.B. Savy, R.W. Mensing, J.C. Chen and B.C. Davis, Seismic Hazard Characterization of the Eastern United States, Vol. 1, "Methodology and Results for Ten Sites," Lawrence Livermore National Laboratory, UCID-20421, April 1985.

11. This matter is under consideration but has not yet received NRC sponsorship (private communication, D.L. Bernreuter, Sept. 1986).

2. S.L. Algermissen, D.M. Perkins, P.C. Thenhaus, S.L. Hangen and B.L. Bender, Probabilistic Estimates of Maximum Acceleration and Velocity in Rock in the Cartiguous United States, U.S. Geological Survey, open file report 821033 (1982).

13. E.g., see R.A. Kerr, "Charleston Quakes are Larger or Widespread," Science, 233, p. 1154, September 12, 1986; and R.A. Kerr, "Eastern Quakes Pinned Down?," Science, 227, January 25, 1985.

14. S.F. Obermeier, G.S. Gohn, R.E. Weems, R.L. Gelinas and M. Rubin, "Geologic Evidence for Recurrent Moderate to Large Earthquakes Near Charleston, South Carolina," Science, 227, pp. 408-411, January 25, 1985.

15. E.g., see P. Talwani and J. Cox, "Paleoseismic Evidence for Recurrence of Earthquakes near Charleston, South Carolina;" Science, 229, pp. 379-381, July 26, 1985.

16. R.E. Weems, S.F. Obermeier, M.J. Pavich, G.S. Gohn, M. Rubin, R.L. Phipps and R.B. Jacobson, "Evidence for Three Moderate to Large Prehistoric Holocene Earthquakes Near Charleston, SC," Proceedings of the Third U.S. National Conference on Earthquakes Engineering, August 24-28, 1986, Charleston, South Carolina, pp. 3-14, (published by the Earthquake Engineering Research Institute).

17. S.F. Obermeier, R.B. Jacobson, B.S. Powars, R.E. Weems, B.C. Hallbick, G.S. Gohn and H.W. Markewich, "Holocene and Late Pleistocene (?) Earthquakes Induced Sand Blows in South Carolina," Proceedings of the Third U.S. National Conference on Earthquake Engineering, August 24-28, 1986, Charleston, SC, pp. 197-208 (published by the Earthquake Engineering Research Institute).

18. S.F. Obermeier, private communication, Sept. 26, 1986.

19. R.M. Thorson, W.S. Playton and L. Seeber, "Geological Evidence for a Large Prehistoric Earthquake in Eastern Connecticut," Geology, 14, pp. 463-467, June 1986.

20. D.L. Bernreuter, private communication, Sept. 1986.

21. A proposal for such studies has been submitted to the NRC by the Structural Analysis Group of BNL.

22. E.g. see Figs. 12 and 13, pp. 3-22 and 3-23, of Ref. 24.

23. R.P. Kennedy, C.A. Cornell, R.D. Campbell, S. Kaplan and H.F. Perla, "Probabilistic Seismic Safety Study of an Existing Nuclear Power Plant," Nuclear Engineering and Design, 59, (1980) 315-338.

24. L.E. Cover, M.P. Bohn, R.D. Campbell and D.A. Wesley, Handbook of Nuclear Power Plant Seismic Fragilities, prepared for the U.S. Nuclear Regulatory Commission by Lawrence Livermore National Laboratory, NUREG/CR-3558 (UCRL-53455), June 1985.

25. D.D. Orvis, C. Johnson and R. Jones, Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment, prepared by NUS Corporation for the Electric Power Research Institute, EPRI NP-3365, Feb 1984.

26. S.H. Bush, "Probability of Damage to Nuclear Components Due to Turbine Failures," Nuclear Safety, 14, No. 3, May-June, 1973.

27. U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, IE Bulletin No. 84-03: "Refueling Cavity Water Seal," August 24, 1984. (This was subsequently followed up by more detailed instructions for evaluation: "Inspection Requirements for IE Bulletin 84-03, "Refueling Cavity Water Seals," Temporary Instruction 2515/66, Inspection and Enforcement Manual, USNRC, Office of Inspection and Enforcement, issue date 12/17/84.)

28. The total count was based on licensee responses to IE Bulletin No. 84-03. As of mid-1985 a few licensees had not yet filed responses.

29. Response of Baltimore Gas and Electric to IE Bulletin No. 84-03.

30. B.J. Garrick et al., "The Effect of Human Error and Static Component Failure on Engineered Safety System Reliability," Holmes & Narver, Inc., HN-194, November 1967.

31. A.D. Swain and H.E. Guttmann, Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, prepared for the U.S. Nuclear Regulatory Commission by Sandia National Laboratories, NUREG/CR-1278 (SAND80-0200), August 1983.

Table 2.1 Typical Spent Fuel Pool Dimensions and Water Inventories

Length/Width/Depth (feet)	Pool Volumes (cubic feet)	Nominal Water Inventory (cubic feet)
40/26/39[a]	4.1×10^4	3.5×10^4
43/22.25/40.25[b]	3.4×10^4	3.3×10^4

[a]BWR, Vermont Yankee.
[b]PWR, Ginna.

Table 2.2 Decay Heat as a Function of Time Since Last Refueling (Data from Appendix A)

Plant	Decay Heat Load (10^6 Btu/hour)			
	Decay Time Since Last Shutdown for Refueling			
	30 days	90 days	0.5 years	1.0 year
Millstone-1	4.43	3.10	2.38	1.76
Ginna	2.62	1.96	1.59	1.25

Table 2.3 Examples of Thermal-Hydraulic Transient Parameters, Assuming Complete Loss of Pool Coolant Circulation

Docket No.[a]	Rate of Temp. Increase (°F/hr)	Time of Boiling[b] (hours)	Boil-Off Rate (gpm)	Boil-Off Rate (ft^3/hr)
50-325	5.0	13.5	28	262
50-250	9.7	9.3	N.A.	-
50-271	<3	>20	14	131
50-247	13.0	4.8	57	534
50-344	<6.3	>11	34	318

[a]See Ref. 1.
[b]Hours after complete loss of cooling capability.

Table 2.4 Fragility Parameters Assumed in This Study for Spent Fuel Storage Pools

Structure	A (g)	β_R	β_U	Ref.
Oyster Creek Reactor Building[a]	0.75	0.37	0.38	24
Zion Auxiliary Building Shear Walls (N-S motion)[b]	1.1	0.12	0.20	25

[a]Designated as the "Kennedy" fragility curves in the text.
[b]Designated as the "Zion" fragility curves in the text.

Table 2.5 Weighting Factors Assigned to the Various Hazard and Fragility Curves for the Millstone Case

Seismic Hazard Curves:	MILLSTONE ω_i	GINNA ω_i
"Best Estimate"	0.50	1.00[a]
15% Confidence Curve	0.10	--
Median Curve	0.30	--
85% Confidence Curve	0.10	--
$\sum\omega_i$ =	1.00	1.00
Fragility Curves:	ω_j	ω_j --
"Kennedy", Median	0.45	0.15
" , 16%	0.15	0.05
" , 84%	0.15	0.05
"Zion", Median	0.15	0.45
" , 16%	0.05	0.15
" , 84%	0.05	0.15
$\sum\omega_j$ =	1.00	1.00

[a]"Synthesized" Curve.

Table 2.6 Events in Which Inflated Seals Have Failed

Date	Plant	Seal Location	Cause	Total Leakage
9/72	Pt. Beach[1]	Transfer Gate	Failure of air supply	11,689 gal.
10/76	Brunswick 2	Inner Pool Gate	Air leak in seal plus compressor power supply failure	(Pool level dropped 5")
5/81	Arkansas Nuclear One - 2	Transfer Gate	Maintenance error, air supply shutoff	1000 gal/min
8/84	Haddam Neck	Cavity Seal	Design weakness, seal shifted	200,000 gal. in 20 min.
10/84	San Onofre 2[1]	Gate Seal	Air compressor power failure	20,000 gal.
11/84	San Onofre 2[1]	Cavity Seal[2]	Manufacturing defect, seal rupture	----
12/86	Hatch	Pool-Canal Flexible Joint	Valve to compressed air supply closed	141,000 gal.

[1]No spent fuel was in the storage pool.
[2]Failure occurred during installation and leak testing.

Table 2.7 Estimated Distribution of Human Error in Heavy Crane Operations. These Estimates, Made by BNL Staff, are Based on Engineering Judgement and are Not Supported by Actuarial Data.

Operational Step	Possible Human Errors	Estimated Fraction of Total Error Frequency[a] (Per Cent)
1. Install rigging	Wrong slings (e.g., hoist rigging not qualified for task)	10
	Improper installation (shackle, pins, etc.)	10
2. Positioning of crane over load, apply tension	Crane hook not over center of gravity (load upset as tension applied)	15
3. Lift load	Control error (wrong hoisting speed unintentional reversal of direction)	10
4. Start horizontal travel	Control error (move wrong direction, lift or lower instead of move)	10
5a. Horizontal travel	Control error (unintentional reversal of motion, overshoot stopping point)	
	Load not high enough to clear obstacles	4
5b. Horizontal travel	Control error or delayed rigging failure resulting in load drop	1
6. Lower load	Control error (wrong direction, descent too fast)	10
7. Positioning of crane over receiving cradle and set-down load	Inaccurate positioning cradle capsizes during set-down	20
	Set down too rapid	10

[a]It is assumed that the movement of a spent fuel shipping cask is carried out by a qualified rigging crew consisting of a foreman, two or more riggers, and a crane operator. The foreman and riggers check each step and crane movements are signaled to the operator by the foreman who stands in a location providing adequate surveillance of the load, and can be clearly seen by the operator.

Table 2.8 Assumptions Used in Calculating the Hazard of Catastrophic Structural Damage to Pool Resulting from the Drop of a Shipping Cask

Item	Assumed Value	Uncertainty Range
Number of fuel shipments (eventual rate to reduce accumulated inventory) per week	2	--
Number of passes over pool edge per shipment	2[a]	0
Fraction of horizontal movement when cask is above pool edge	0.25	0.1 to 0.5
Total operational time in each movement, minutes per lift	10	8 to 30
Time over pool edge per lift, seconds per lift	10	5 to 20
Mechanical failure rate of crane, per operating hour	3×10^{-6}	10^{-6} to 10^{-5}
Electrical failure rate of crane, per operating hour	3×10^{-6}	10^{-6} to 10^{-5}
Total accident rate from human error, failures per lift	6×10^{-4}	10^{-4} to 10^{-3}
Fraction of human error cask drop accidents occurring during horizontal motion of crane, fraction of total	0.01	5×10^{-3} to 5×10^{-2}
Conditional probability of structural failure of pool given a cask drop at pool edge location, failures per drop	0.1	10^{-2} to 1.0

[a]Some spent fuel pools have a special section for the shipping cask separated from the main pool by a wall with a wier or gate. For such a configuration the number of passes over the "pool edge" would be zero and hence the risk to the main pool from a cask drop would be zero.

Table 2.9 Summary of Estimated Probabilities for Beyond Design Basis Accidents in Spent Fuel Pools Due to Complete Loss of Water Inventory

Accident	Estimated Probability/Ry	
	Millstone	Ginna
Loss of Pool Cooling Capability	1.4×10^{-6}	5.7×10^{-7}*
Seismic Structural Failure of Pool	2.2×10^{-5}	1.6×10^{-5}
Structural Failure from Tornado Missiles	$<1 \times 10^{-8}$	$<1 \times 10^{-8}$
Structural Failure from Aircrash	$<1 \times 10^{-10}$	$<1 \times 10^{-10}$
Structural Failure from Turbine Missile	4×10^{-7}	-0**
Loss of Pool Water Due to Pneumatic Seal Failure	0	1×10^{-6}
Structural Failure from Cask Drop[1]	3.1×10^{-5}	3.1×10^{-5}

[1]After removal of accumulated inventory resumes.

*With credit for third cooling system. Other PWRs which typically have two spent fuel cooling systems would have an estimated fuel uncovery frequency of about 1×10^{-6}/Ry.

**Typical PWRs may have a failure frequency due to turbine missiles on the order of 4×10^{-7} but Ginna's pool is shielded from the turbine.

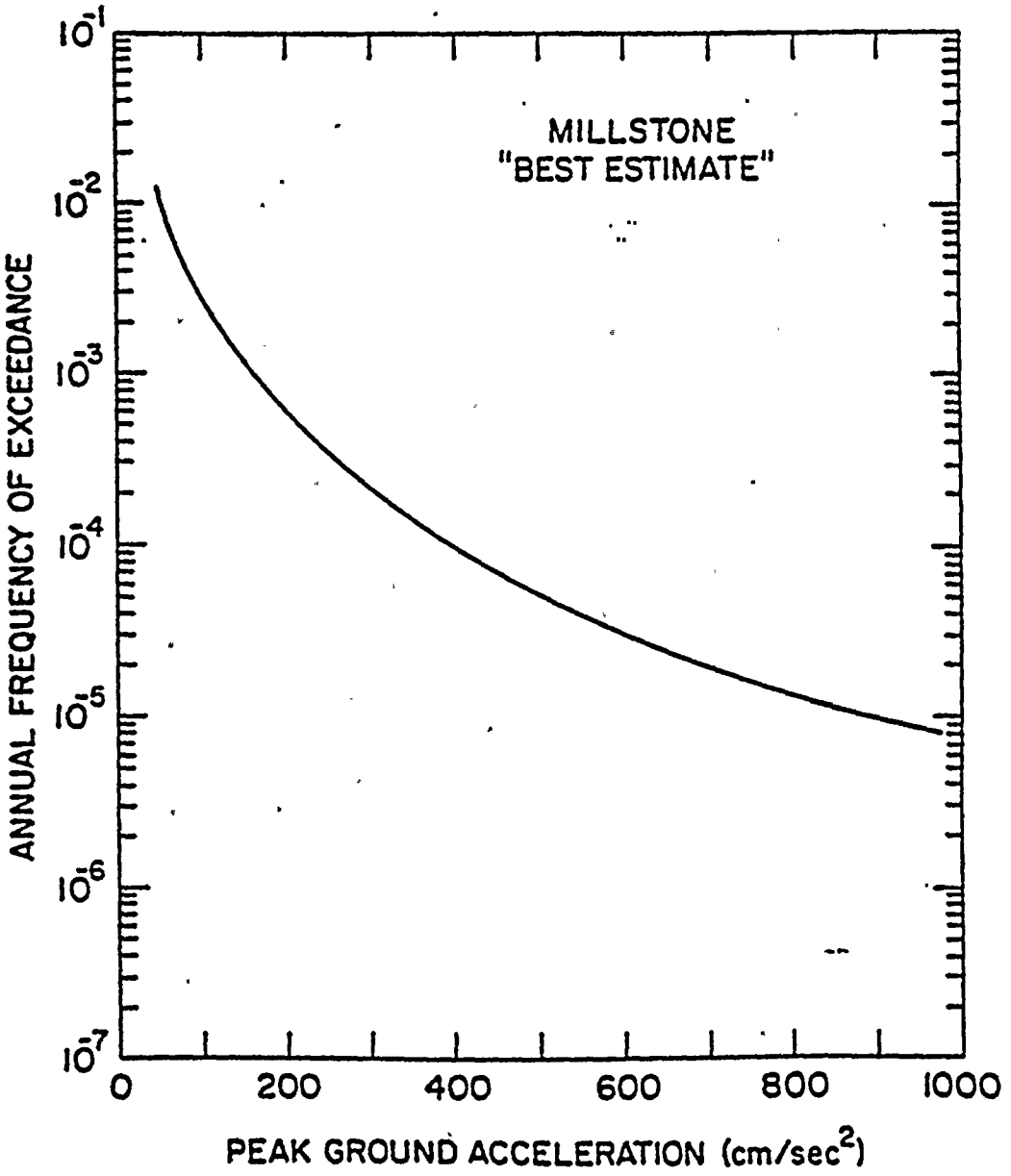

Figure 2.1. Seismic Hazard Curve for the Millstone Site. The curve shown is the mean of the hazard curves generated from the "best estimate" input data of the ten experts participating in the SHC study combined with the "best estimate" model of the ground motion panel. Site corrections are included (Source: Ref. 10, pg. 5-43).

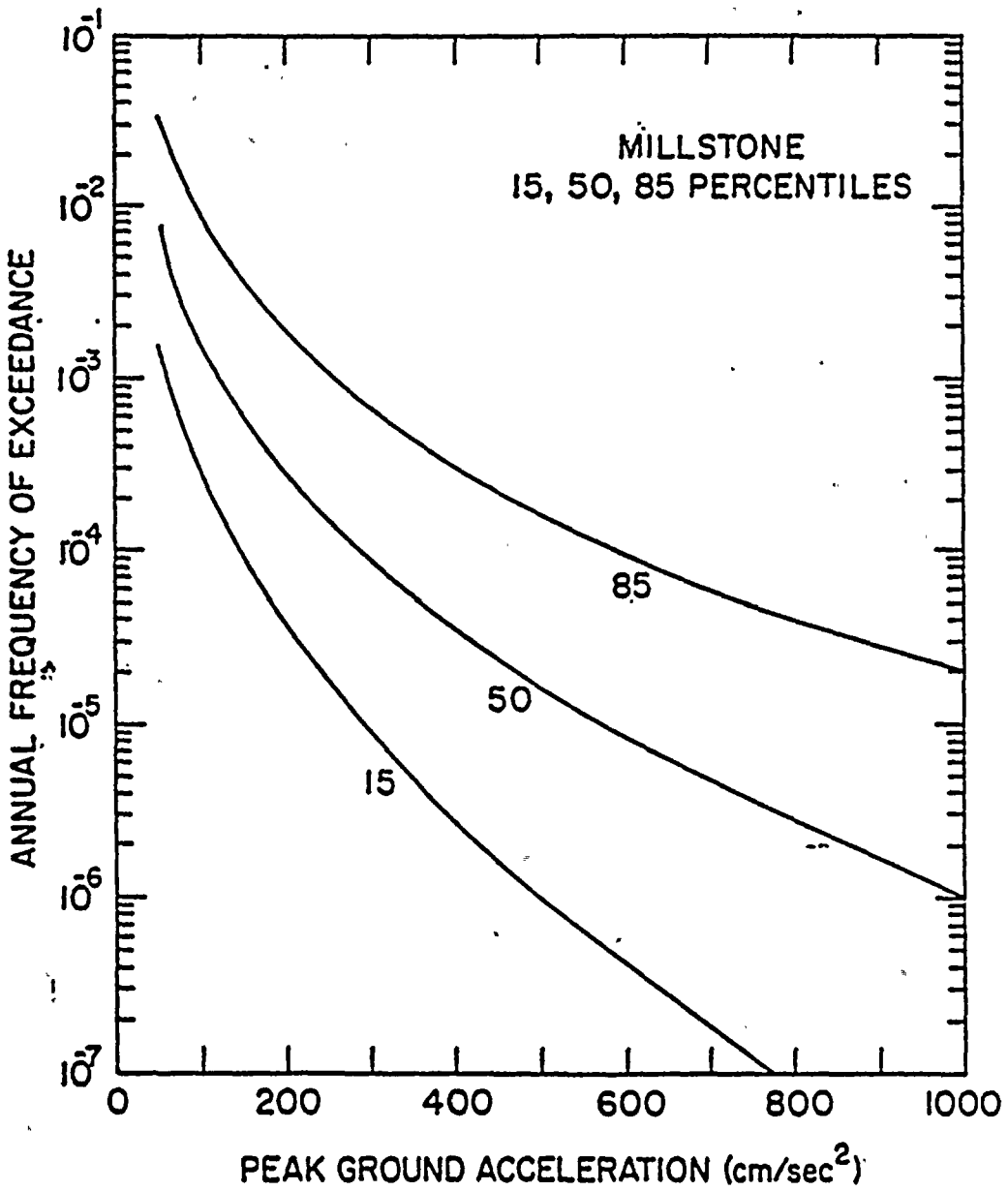

Figure 2.2. The 15, 50 and 85 Percentile Hazard Curves for the Millstone Site. The data are based on confidence levels in the input seismicity data of the experts and uncertainties in the best choice of ground motion models (Source: Ref. 10, pg. 5-45).

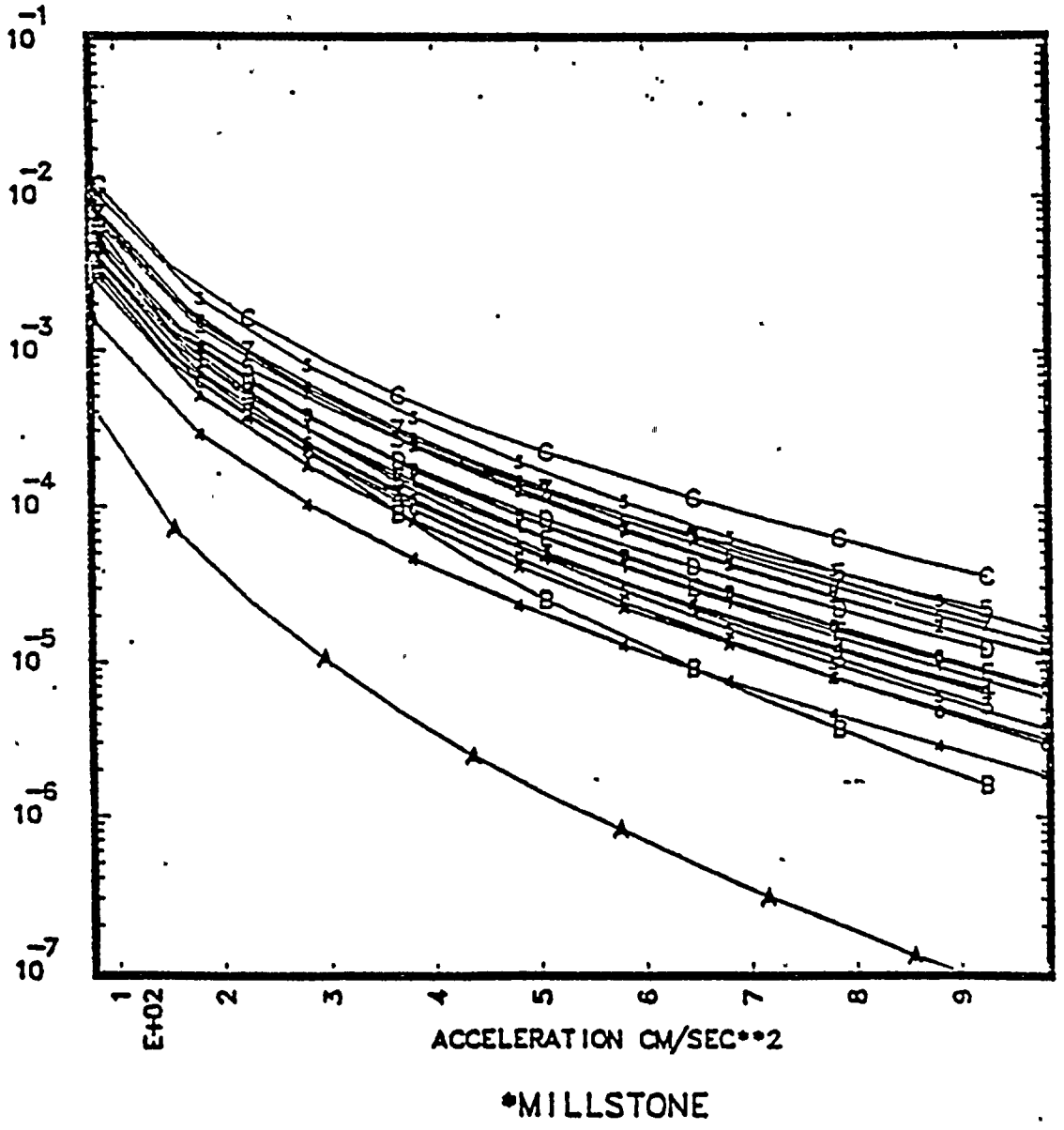

Figure 2.3. Seismic Hazard Curves for Millstone of Each of the Individual Experts Participating in the SEP Studies (Ref. 8) and/or the SHC Studies (Ref. 10). The curves give an indication of the spread in expert opinion (Source: Ref. 10, pg. 209).

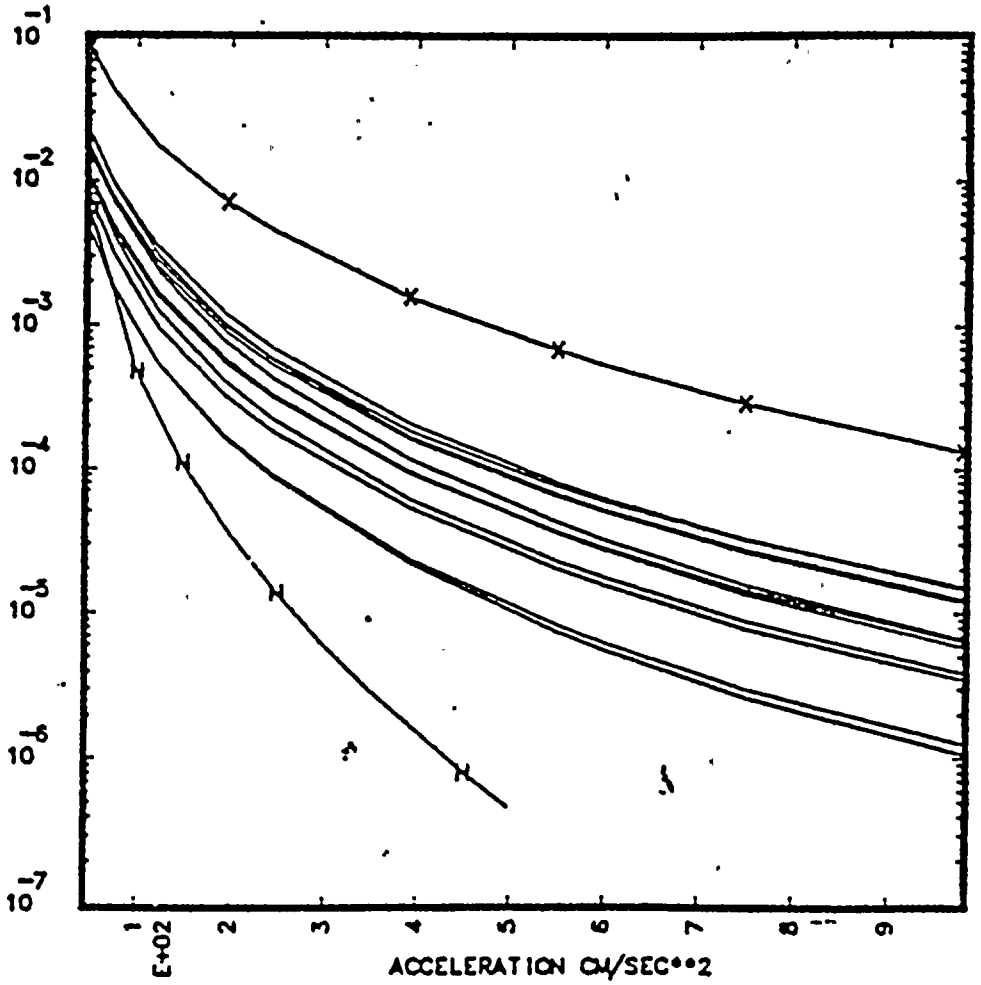

Figure 2.4 Comparison of the Millstone Site Hazard Curves Generated from the Data Input of the SHC Experts, with Those Generated from the USGS Data (Curve "X") and from the Historical Record of the Past 280 Years (Curve H) (Source: Ref. 10, pg. 6-7).

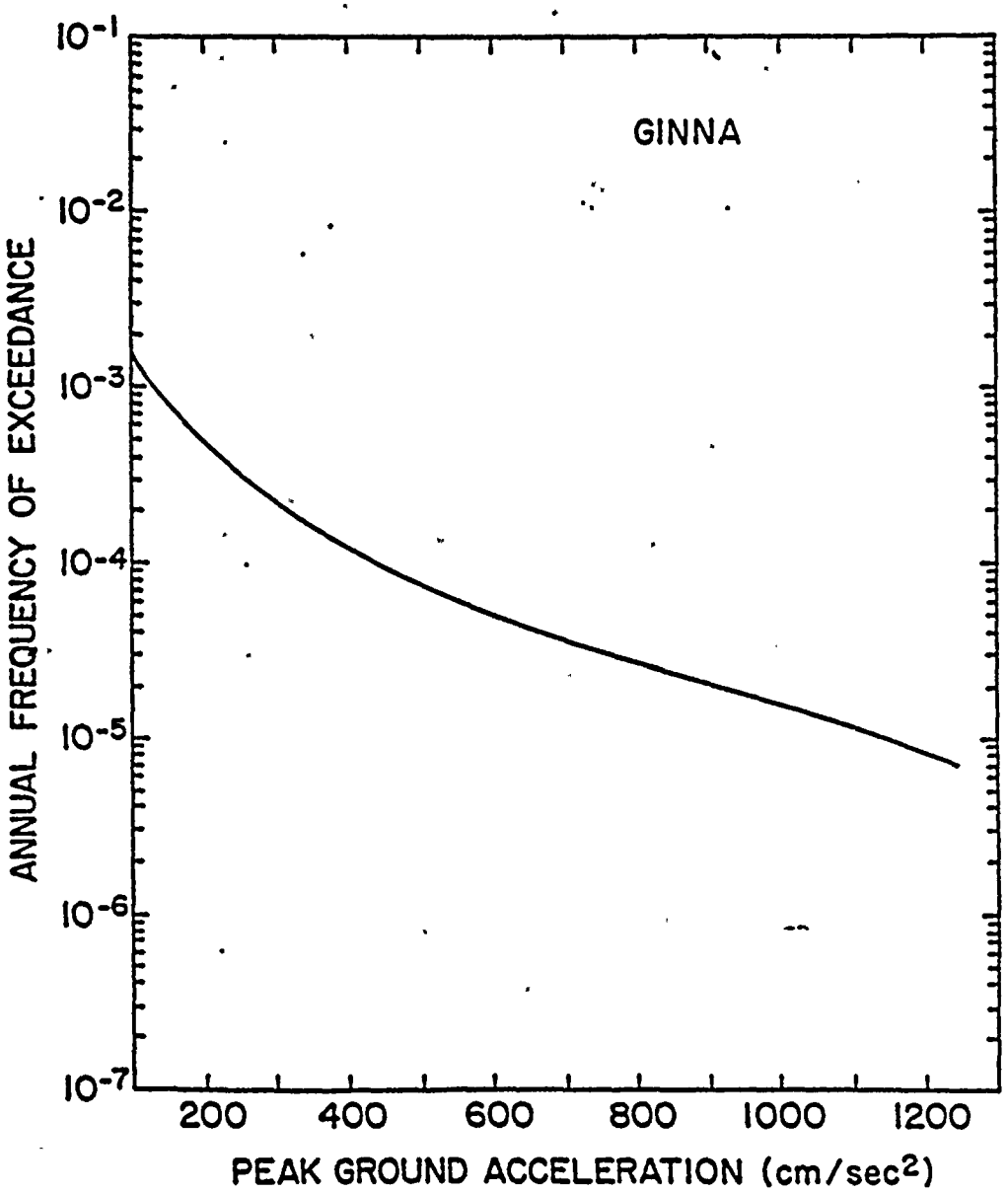

Figure 2.5 Seismic Hazard Curve for Ginna. This curve was Synthesized from the SHC "Best Estimate" curve for Millstone (see Figure 2.1), and PGA ratios for Millstone and Ginna given in the SEP studies.

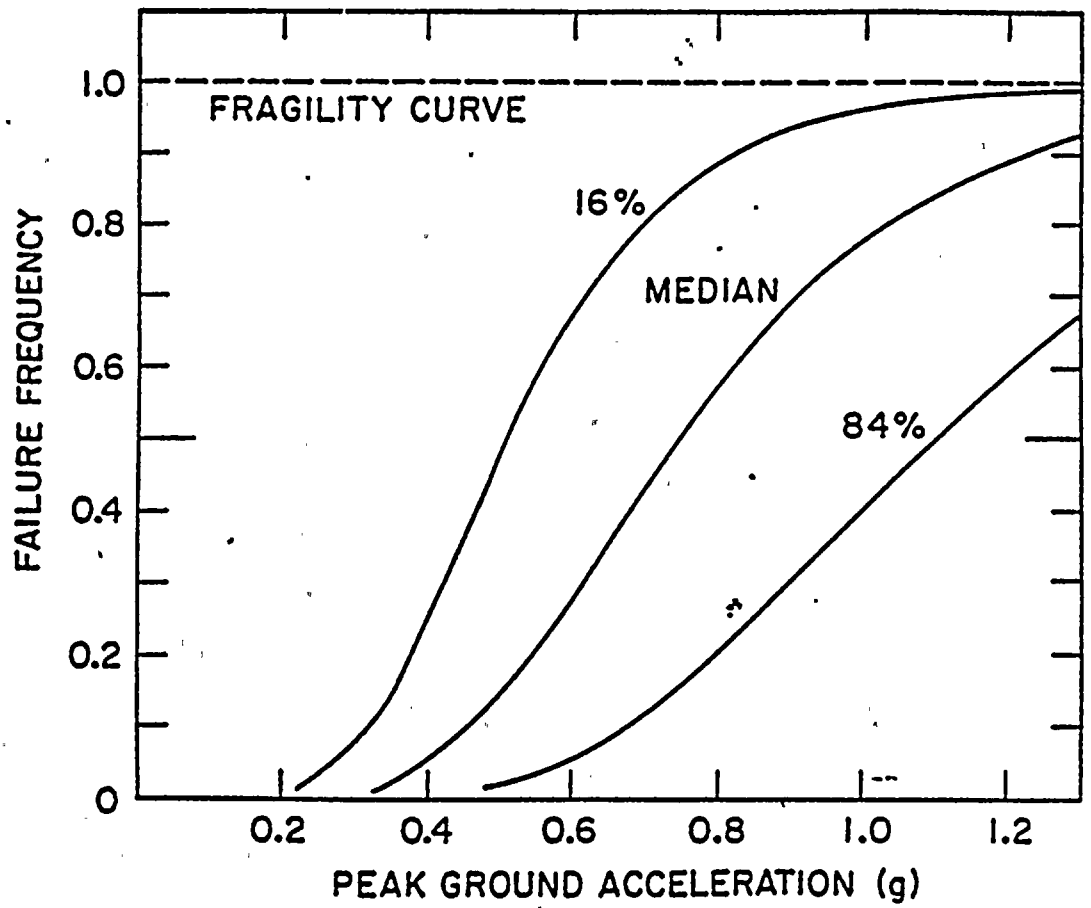

Figure 2.6 Fragility Curves for the Oyster Creek Reactor Building. The curves generated by R.P. Kennedy et al. (Ref. 23) give the frequency of structural failure as a function of peak ground acceleration during an earthquake.

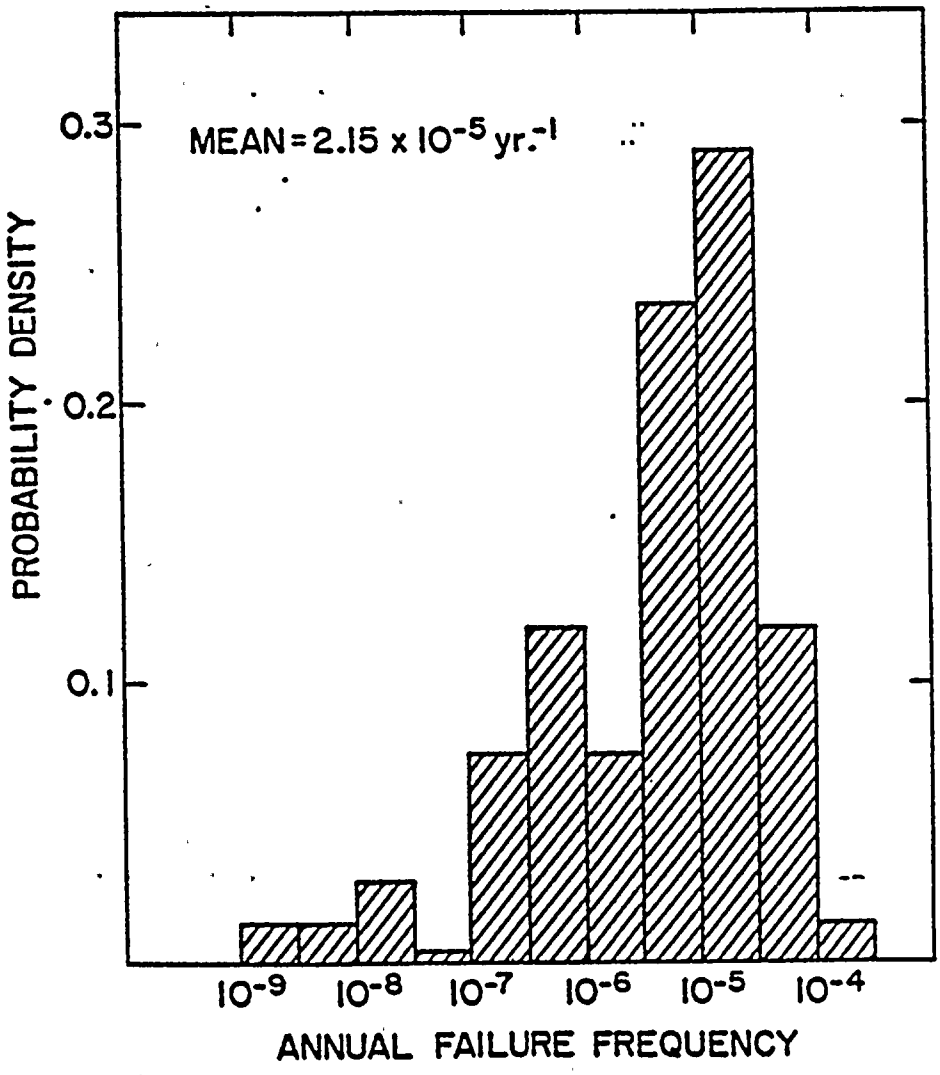

Figure 2.7 Probability Density for Seismically-Induced Failure as a Function of Annual Failure Frequency. The histogram was obtained from 24 convolutions of four hazard curves with six fragility curves and includes the weighting factors assigned to each curve.

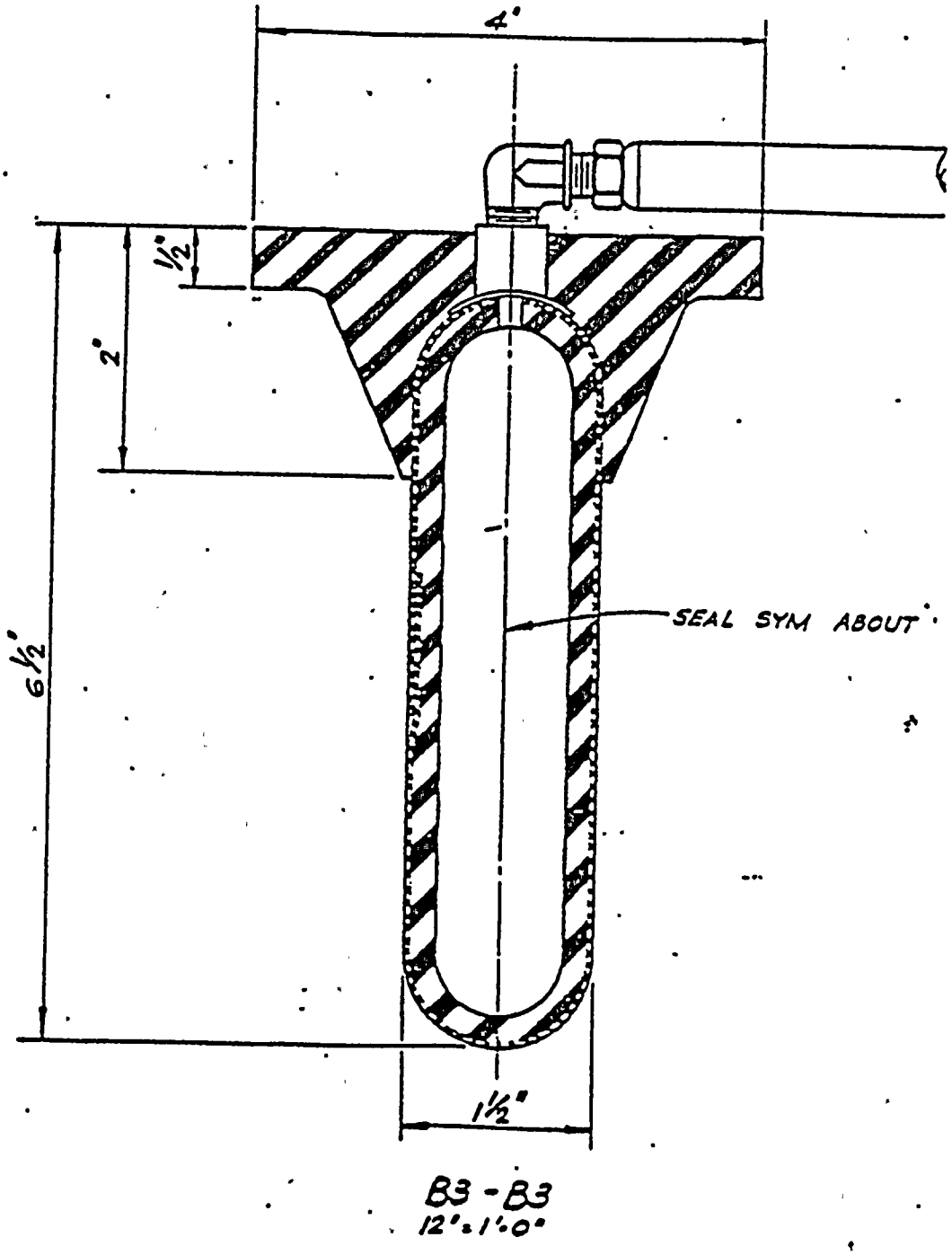

Figure 2.8 Cross Section of a Typical Pneumatic Seal (Source: submission by Sequoyah Nuclear Plant, Docket No. 50-327, 10/26/84 in response to IE Bulletin 84-03.

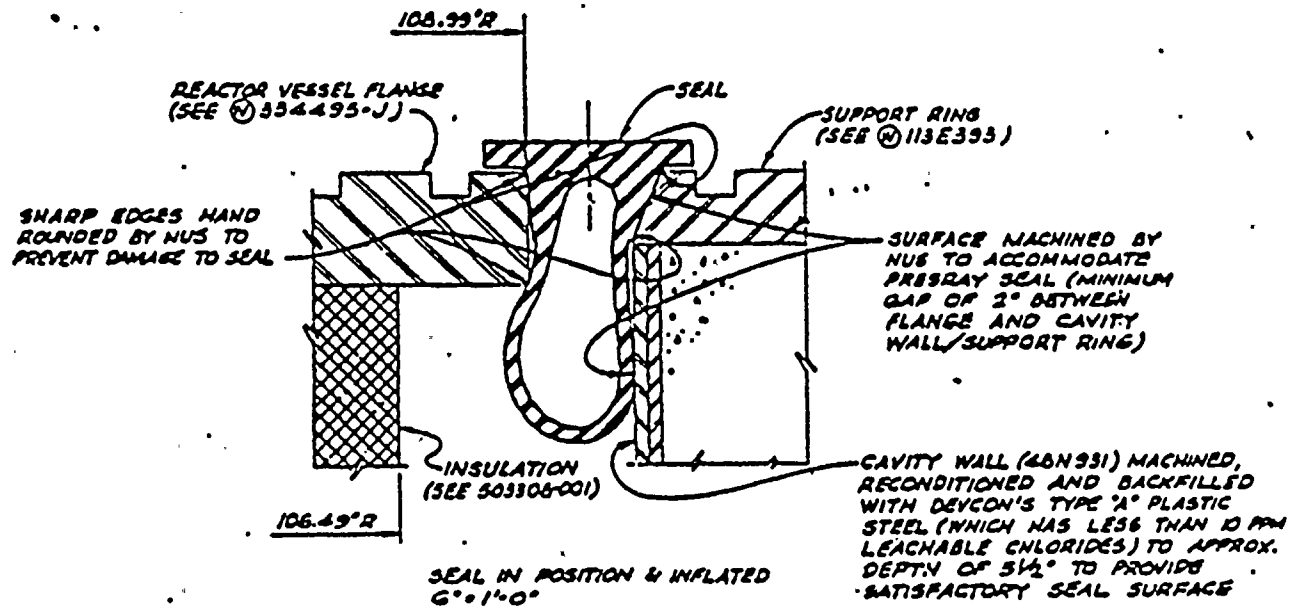

Figure 2.9 Cross Section of Inflated Pneumatic Seal Seated on the Reactor Vessel Flange and Inner Surface of Cavity Wall (Source: submission by Sequoyah Nuclear Plant, Docket No. 50-327, 10/26/84 in response to IE Bulletin 84-03.

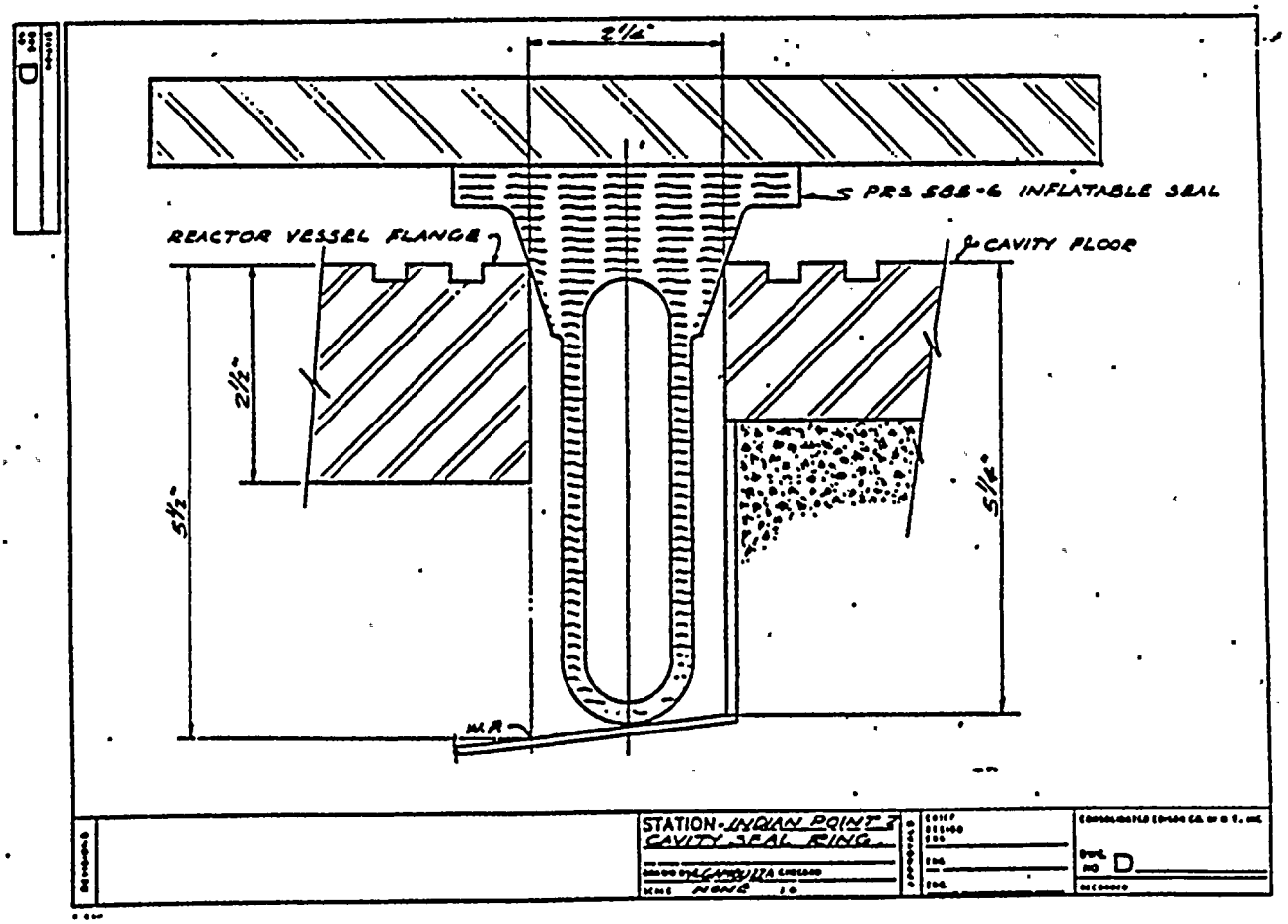

Figure 2.10 Uninflated Pneumatic Seal with Steel Hold-down Ring (Source: submission by Indian Point Station, Unit 2, Docket No. 50-247, 11/30/84 in response to IE Bulletin 84-03).

3. EVALUATION OF FUEL CLADDING FAILURE

Two previous studies[1,2] have analyzed the thermal-hydraulic phenomena assuming a complete drainage of the water from a spent fuel pool. The previous section addressed the possible mechanisms for such an accident and provided estimates for the accident frequency. This section provides a reevaluation of the basis for the SNL results.[1,2]

3.1 Summary of SFUEL Results

The SFUEL code was developed by Benjamin et al.[1] to analyze the behavior of spent fuel assemblies after an accident has drained the pool. The results reported in Reference 1 indicated a wide range of decay power levels for which self-sustaining oxidation of the cladding would be predicted. Several limitations in the SFUEL model were identified and addressed in a subsequent investigation.[2] But comparisons to small scale experiments were not very successful.

3.1.1 Summary Model Description

The SFUEL code was developed at SNL and is described in Reference 1. Basically it is a finite difference solution of the transient conduction equation for heating of the fuel rods considering:

- The heat generation rate from decay heat and oxidation of the cladding.
- Radiation to adjacent assemblies or walls.
- Convection to buoyancy-driven air flows.

The key assumptions in the analysis are:

1) The water drains instantaneously from the pool.

2) The geometry of the fuel assemblies and racks remains undistorted.

3) Temperature variations across the fuel rods are neglected.

4) The air flow patterns are one-dimensional.

5) The spaces between adjacent basket walls are assumed to be closed to air flow.

After the water is drained from the pool the fuel rods heat up until the buoyancy driven air flow is sufficient to prevent further heating. If the decay heat level is sufficient to heat the rods to about 900°C, (1650°F) the oxidation becomes self-sustaining. That is the exothermic oxidation reaction provides sufficient energy to match the decay heat contribution and the temperature rises rapidly.

Reference 2 modified the SFUEL code to increase calculational stability and assess propagation of Zircaloy "fires" from high power to low power assemblies. The SFUEL code was also modified by Pisano et al.[2] to eliminate unrealistically high temperatures* by non-mechanistically removing each node as it reaches the melting point of Zircaloy dioxide (2740°C or 4963°F). In the present investigation, the oxidation cutoff has been reduced to 1900°C (The melting point of Zircaloy 3450°F).

3.1.2 Clad Fire Initiation Results

An extensive review of the cladding oxidation models used in SFUEL,[1,2] is given in Appendix B:

1. The likelihood of clad fire initiation is not very sensitive to the oxidation equation.

2. The oxidation equation used in SFUEL is a reasonable representation of the data.

*Since the code does not explicitly treat melting of the cladding, temperatures as high as 3800°C were predicted.[2]

3. The likelihood of clad fire initiation is most sensitive to the decay heat level and the storage rack configuration (which controls the extent of natural convection cooling).

The critical conditions for clad fire initiation are summarized in Table 3.1. Note that for the old style cylindrical fuel racks with a large inlet orifice (3 inch diameter) the natural convection cooling in air is predicted to be adequate to prevent self-sustaining oxidation (cladding "fires") after 10 days of decay for BWR assemblies and 50 days for PWR assemblies. However for the new high density fuel racks, natural convective flows are so restricted that even after cooling for a year there is potential for self-sustaining oxidation. As pointed out by Benjamin et al.[1] there are a number of modifications to the fuel rack design which would enhance convective cooling and reduce the potential for cladding fires. However, the limited flow area of the high density designs make it difficult to ensure adequate cooling by natural convection of air.

For the assumption of annual discharges, the critical decay time can be translated into a likelihood of cladding fire for a complete loss of pool water inventory. For the critical cooling times given in Table 3.1 the probability of self-sustaining oxidation is approximately:

0.0 to 0.5 for BWRs with low density storage racks,

0.0 to 0.7 for PWRs with low density storage racks, and

1.0 for PWRs with high density storage racks.

3.1.3 Clad Fire Propagation

The SNL investigations[1,2] of spent fuel behavior after a loss of pool integrity accident (assumed to result in complete drainage of the pool), identified a range of power levels necessary for the initiation of self-sustaining clad oxidation and substantially lower power levels at which adjacent fuel bundles would oxidize once oxidation had been initiated. However, the phenomenology of propagation is not well understood and there is considerable

uncertainty in these estimates. Benjamin et al.,[1] Pisano et al.,[2] Han[3] and Johnsen[4] have pointed out a number of limitations in the previous analyses.[1,2] In order to put the present results in perspective it is worth mentioning the most important limitations:

1. The oxidation equation allows oxidation to continue beyond 1900°C (3450°F) where clad melting and relocation is expected. PBF and KfK tests show clad relocation at temperatures in the range of 1900° to 2200°C but the analyses have calculated temperatures as high as 3500°C (6330°F) without accounting for clad and fuel melting. At such high temperatures the radiation heat flux becomes very large and it is believed that the potential for propagation to adjacent bundles will be overestimated.

   In order to provide more realistic estimates of the potential for oxidation propagation, BNL has chosen to terminate oxidation at the Zircaloy melting point.

2. The SFUEL code had not yet been validated successfully against fuel rod oxidation data. A preliminary comparison[2] against SNL data was only partially successful.

   The SFUEL code has been compared to the SNL data in a separate section (3.2) and key portions of the code have been validated. Specifically, the axial heat up (without oxidation) and the temperature at which self sustaining oxidation is reached has been validated. If a low power spent fuel bundle heats up to within one or two hundred °C of self-sustaining oxidation due to its own internal energy there is a high likelihood that the additional energy from an adjacent high power bundle will be sufficient to bring it to the initiation point.

3. The reaction rate equation has been criticized as being too low for long term exposure at low temperatures (when oxide layers may flake off and expose fresh Zircaloy). However, Appendix B has shown that the SFUEL calculations are not very sensitive to the low temperature oxidation rate.

4. The lack of a fuel and clad melting and relocation model has also been criticized.

   Development of realistic degraded fuel models is beyond the scope of the present investigation. However, we believe that the modified SFUEL code (SFUEL1W[2]) has sufficient flexibility to estimate the importance of oxidation propagation.

5. Johnson[4] criticized the clad failure criterion used in the SNL analyses.[1,2] He noted that the clad failure could occur at temperatures as low as 650°C if the thermal loading is sustained for several hours.

   In view of the large uncertainty in the thermal behavior, we agree that a prediction of temperatures in excess of 650°C should not be viewed as successful cooling of the assembly. At these temperatures cladding failure and fission product release is very likely and the potential for cladding "fires" is high due to the effects of asymmetric heating (from adjacent high power bundles).

Propagation of cladding "fires" by particulate (i.e., spallation) or zirconium vapor transport has been investigated and eliminated in an approximate separate effects study by Pisano et al.[2] However, propagation due to the heat flux (radiation and convection) from adjacent bundles is predicted to occur even to very low power assemblies (at power levels corresponding to 3 years of decays).

The purpose of this section is to establish the range of conditions for which propagation is predicted to occur. Both the power of the initiating bundle and the power of the adjacent bundles have been varied as well as the ventilating conditions of the spent fuel building.

It should be emphasized that SFUEL does not address the question of Zircaloy oxidation propagation after clad melting and relocation. For recently discharged fuel (less than 90 days), or for severely restricted air flow (e.g., high density PWR spent fuel racks) the oxidation reaction is predicted

to be very vigorous and failure of both the fuel rods and the fuel rod racks is expected. Thus a large fraction of the fuel rods would be expected to fall to the bottom of the pool forming a large debris bed. If water is not present in the bottom of the pool, the debris bed will remain hot and will tend to heat the adjacent assemblies from below. The investigation of debris bed formation is beyond the scope of the present study, but it appears to be an additional mechanism for oxidation propagation.

Results

As pointed out in Section 3.1.1 self-sustaining oxidation initiation is not very sensitive to the oxidation rate equation but it is dependent upon the calculated air flow (related to flow resistance) and the power level. BWR spent fuel with its low power density and open flow configuration must be recently discharged (within about 3 months) for self-sustaining oxidation to be initiated and unless it is a very high power bundle (discharged within 10 days or less) there is only a slight chance of propagation to older low power fuel bundles.

However, PWR spent fuel racks typically have a higher power density and more flow restriction, thus self-sustaining oxidation can be initiated in fuel that has been discharged for one year or more.

Two fuel building ventilation conditions have been investigated as described below but it must be recognized that both of these assumptions correspond to very idealized conditions that are unlikely to be duplicated in an actual accident. Rather these idealized conditions are provided to demonstrate the importance of the various assumptions. For a beyond design basis seismic event, that catastrophically fails the pool, it seems likely the failure of the fuel building may also occur. However, Benjamin et al.[1] have shown that a very large hole (at least 77 ft^2) must be opened in order to approximate the perfect ventilation case.

Perfect Ventilation

Under the perfect ventilation condition it is assumed that the fuel building is maintained at ambient conditions by a high powered ventilation system (note that the flow rate must be much higher than typical gas treatment systems) or by a large opening (greater than 77 ft^2) in the building. Oxygen is not depleted and the air entering the pool is assumed not to be heated by the hot gases exiting the fuel assemblies. The conditions necessary to initiate self-sustaining oxidation under perfect ventilation conditions were summarized in Table 3.1 for three typical fuel rack configurations. Note that these are "borderline" conditions in that a slightly lower power level or a larger inlet hole size would be predicted to prevent self-sustaining oxidation from occurring. Note that the "critical" conditions outlined in Table 3.1 do not imply that fuel rod failure is not predicted for power levels below these conditions. The power level must be reduced substantially (about 20%) to ensure that the predicted clad temperature is below 650°C (the minimum temperature at which clad failure and fission product release is likely[5] to occur).

For power and flow conditions that are only slightly below the "critical" conditions it should be obvious that the heat flux from a much higher power adjacent bundle would have the potential to push the "non-critical" fuel over the self-sustaining oxidation threshold. Thus the only real propagation question is whether recently discharged (high power) spent fuel will radiate sufficient energy to initiate self-sustaining oxidation in low power fuel bundles that have been cooled for one or more years.

In this context two limitations of the SFUEL1W[2] code should be noted:

1. The fuel storage racks are assumed to be immediately adjacent so that no air flow between racks is allowed. (The numerical approach used to calculate the heat transfer is numerically unstable if flow is allowed).

2. All fuel storage racks are assumed to be identical so that the question of propagation from high power cylindrical racks to low power high density racks cannot be addressed.

The first limitation probably represents current storage practices where a number of fuel pools are approaching their design capacity. However, the question of providing deliberate cooling channels between recently discharged fuel and the older fuel cannot be directly addressed. Based on engineering insight, it appears that, under the idealized perfect ventilation conditions, the provision of an air space of 6 to 12 inches around the periphery of recently discharged fuel would minimize the likelihood of oxidation propagation to low power spent fuel assemblies. (Note that the code does allow for an air space adjacent to the pool walls and 6 to 12 inches is found to be adequate if flow through the bundles is not restricted.)

Since high density fuel storage racks are predicted to cause self-sustaining oxidation even after storage for one or more years, it seems clear that it would be undesirable to store spent fuel in high density storage racks if it has been discharged within the last two years. (It may be worth noting that current practice restricts the storage density of low burnup fuel due to nuclear criticality considerations.) Thus the question of propagation from cylindrical fuel racks to high density fuel racks should be addressed, but the second limitation mentioned above precludes intermixing of the storage rack configurations.

The propagation results with perfect ventilation are summarized in Table 3.2 for the high density rack configuration described in Reference 2. Note that the lowest power (11.0 kW/MTU) for self-sustaining clad oxidation corresponds approximately to fuel that has been discharged for one year, but the oxidation reaction will generate sufficient energy to propagate to a fuel bundle that is about 2 years old (6.0 kW/MTU). For a fuel assembly that has been discharged for about 10 days (90 kW/MTU) the high decay heat level causes extensive clad oxidation in the high power bundle and a somewhat higher propensity to propagate to low power fuel assemblies (as low as 5 kW/MTU which corresponds roughly to a 2-1/2 year old discharge).

The propagation results for a low density fuel rack (cylindrical) with a 3 inch diameter inlet hole is summarized in Table 3.3. Note that the range of power for the high power assembly is limited due to the improved free convection within this type of fuel rack. Thus self-sustaining clad oxidation is

initiated at decay power levels at or above 30 kW/MTU (corresponding to about 90 days of cooling). Assuming that more than one discharge per year is unlikely, the adjacent low power assembly must be less than or equal to about 19 kW/MTU (180 days of cooling). Thus propagation only occurs for fuel that has been discharged less than 1 year with initiation from fuel that has been discharged within 2 weeks.

For a PWR cylindrical fuel rack with only a 1.5 inch diameter flow hole, the air flow is much more restricted and the possibility of propagation is stronger as indicated in Table 3.4. For the 1.5 inch hole size propagation is predicted to occur for cooling times as long as two years.

Inadequate Ventilation

As previously mentioned the case of perfect ventilation implies a very high ventilation rate that is not normally possible. Benjamin et al.[1] extended the SFUEL code to consider limited heat removal to just keep the spent fuel building at constant pressure. Details of the modeling are described in Reference 1, but the main result of the model is that the fuel building atmosphere heats up (due to decay heat and the chemical energy of oxidation) and the oxygen is depleted. Benjamin[1] found that the heat-up of the building increased the likelihood of self-sustaining oxidation (i.e., decreased the decay power level necessary to initiate self-sustaining oxidation). This section is intended to address the question of whether limited ventilation also increases the likelihood of propagation to low power bundles.

Table 3.5 provides a summary of propagation runs under inadequate ventilation conditions. For the analyses the high power assemblies are modeled to represent approximately 1/3 of the core for 1000 MWe plant and the fuel building is taken to have a volume of 150,000 ft^3. The results given in Table 3.5 indicate that propagation is no more likely with inadequate ventilation than with perfect ventilation. In fact propagation does not occur for several conditions listed in Table 3.5 for which propagation was predicted with perfect ventilation. Although this result is somewhat surprising, it is simply a result of the oxygen depletion calculation. That is, the oxidation of the

recently discharged assemblies uses up the oxygen supply before the lower power assemblies can be heated to the point of self-sustaining oxidation.

In view of the potential for fuel building failure due to either the assumed initiating event (e.g., a beyond design basis earthquake) or the rapid building pressurization from Zircaloy combustion and decay heat, BNL considers the oxygen depletion calculation to be unrealistic. Thus, in spite of the many uncertainties, the perfect ventilation model is expected to give the best approximation for the potential for propagation.

Conclusions Regarding Propagation

Based on the previous results we have concluded that the modified SFUEL code (SFUEL1W[2]) gives a reasonable estimate of the potential for propagation of self-sustaining clad oxidation from high power spent fuel to low power spent fuel. Under some conditions, propagation is predicted to occur for spent fuel that has been stored as long as 2 years.

The investigation of the effect of insufficient ventilation in the fuel building indicated that oxygen depletion is a competing factor with heating of the building atmosphere and propagation is not predicted to occur for spent fuel that has been cooled for more than three years even without ventilation.

These results are in general agreement with the earlier SNL studies.[1,2]

3.2 Validation of the SFUEL Computer Code

The SNL investigations[1,2] of spent fuel behavior after a loss of pool integrity accident (assumed to result in complete drainage of the pool), identified a range of power levels necessary for the initiation of self-sustaining clad oxidation and substantially lower power levels at which adjacent fuel bundles would oxidize once oxidation had been initiated. However, an attempt[2] to validate the code was only minimally successful in that the post-test analyses were able to match the heat-up rate in helium (without oxidation) but the SFUEL code over-estimated the temperature transient after air was introduced.

The objective of this section is to use the revised[3] oxidation rate equation in SFUEL to analyze the SNL small scale tests to aid in validating the SFUEL code. The SNL tests are described in Reference 2, but in order to put the test results in perspective several important conditions should be highlighted:

1. The test was of a small bundle of electrically heated rods (9 rods) with a short length (38 cm).

2. In order to achieve self-sustaining clad oxidation (>850°C) the rods were heated with a very low flow rate of helium before air was admitted to the test assembly.

Under these test conditions the dominant heat loss is via radiation whereas for the postulated accident the dominant heat loss is via free convection. These test conditions lead to laminar flow (a Reynolds number of about 100) in which oxygen diffusion to the cladding surface limits the reaction rate. Only one test (6) had a sufficiently high air flow rate to allow vigourous oxidation.

Since the free convection and radiation calculations in SFUEL[1,2] were inappropriate to the test configuration, Pisano et al.[2] created a stripped down version called CLAD[2] which used a matrix inversion routine to calculate radiation losses.

After several preliminary attempts to analyze the helium portion of the tests we concluded that there were several errors which led to underestimation of the convection portion of the heat losses. Since helium has a much higher heat capacity and conductivity than air it appears to contribute to establishing the initial conditions. In order to provide an adequate simulation of the initial steady-state portion of the test we made two modifications to the CLAD code:

1. Include helium properties with a switch to air properties at the start of the transient.

2. Include an energy balance on each gas control volume to force conservation of energy.

With these changes we were able to obtain an adequate simulation of the initial portion of the tests. Using this revised version of CLAD with the Weeks' oxidation correlation,[3] analysis of both the helium and the air portions of the test looked reasonable, but still tended to over-predict the peak temperatures during oxidation. In order to bring the calculations into reasonable agreement with the small scale data the Weeks' correlation has been reduced by a factor of four (note that this corresponds approximately to the data scatter).

Results

The revised CLAD code has been used to analyze the SNL small scale experiments Tests 4, 5 and 6. The other three tests were intended to simulate propagation with nonuniform heating and structures that CLAD was not capable of modeling. The CLAD results for Test 4 are compared to the data in Figure 3.1. These results still tend to overpredict the temperature in the center of the test rod, but give reasonably good agreement at the top of the rod where radiative heat losses are large.

The peak temperatures calculated by CLAD are summarized in Table 3.6 and compared to the peak measured temperatures for the three tests. Note that CLAD still overpredicts the peak temperature for the low flow rate test (4 and 5) but gives good agreement with the high flow rate tests where adequate oxygen is available. It should be noted that this "oxygen starvation" phenomenon appears to be a result of the extremely low laminar flow where oxygen must diffuse to the clad surface. CLAD includes an oxygen depletion calculation but assumes that all the oxygen in each volume is immediately available at the surface.

3.3 Conclusions Regarding SFUEL Analyses

After an extensive review of the SFUEL code and comparison to the SNL small scale experiments, BNL concludes that the code provides a valuable

tool for assessing the likelihood of self-sustaining clad oxidation for a variety of spent fuel configurations assuming that the pool has been drained.

The SNL small scale data provide a reasonable degree of validation for the heat-up and oxidation models, but the results are extremely sensitive to the natural convection calculation which has not been validated.

When oxidation is terminated at the Zircaloy melting temperature (assuming that the molten Zircaloy is relocated), oxidation propagation only occurs for spent fuel bundles which are already approaching the "critical" conditions for self-sustaining oxidation (see Table 3.1). However, this finding does not mean that oxidation propagation is unlikely. On the contrary, for some high density storage configurations the "critical" conditions are approached for spent fuel that has decayed for two to three years. Thus clad "fire" propagation appears to be a real threat but the basic question remains as to what are the "critical" conditions for initiation of oxidation and what the uncertainty is for a given spent fuel configuration. The critical conditions are summarized in Table 3.1 for several typical spent fuel racks. While the heat-up and oxidation models have been validated to a limited extent by the SNL data (see Section 3.2), the authors believe that the largest source of uncertainty is in the natural convection flow rate. It is recommended that these free convection flow calculations be verified against large scale data. Preferably the data would be obtained from spent fuel assemblies in typical storage racks (both high and low density).

3.4 References for Section 3

1. A.S. Benjamin, D.J. McCloskey, D.A. Powers, S.A. Dupree, "Spent Fuel Heat-up Following Loss of Water During Storage," NUREG/CR-0649, March 1979.

2. N.A. Pisano, F. Best, A.S. Benjamin, K.T. Stalker, "The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool," Draft Report, January 1984.

3. J.T. Han, Memo to M. Silberberg, USNRC, May 21, 1984.

4. G.W. Johnsen, Letter to F.L. Sims, EG&G, Idaho, April 4, 1984.

Table 3.1 Summary of Critical Conditions Necessary to Initiate Self-Sustaining Oxidation

Spent Fuel Rack Configuration	Inlet Orifice Diameter (inches)	Minimum Decay Power (kW/MTU)	Approx. Critical[1] Decay Time (days)
High Density PWR (6 assemblies per rack)	5	6	700
High Density PWR (6 assemblies per rack)	10	11	360
Cylindrical PWR	5	90	10
Cylindrical PWR	3	45	50[2]
Cylindrical PWR	1.5	15	250[2]
Cylindrical BWR	1.5	14	180
Cylindrical BWR	3.0	70	<10

[1] Critical cooling time is the shutdown time necessary to reach a decay power level below the minimum decay power for self-sustaining oxidation. The cooling time to prevent cladding failure is at least 20% longer.

[2] Note that these critical cooling times are somewhat lower than that found by Benjamin et al.[1] since the orifice loss coefficient was modified at BNL

Table 3.2 Summary of Radial Oxidation Propagation Results for a High Density PWR Spent Fuel Rack with a 10 Inch Diameter Inlet and Perfect Ventilation

High Power Level (kW/MTU)	Adjacent Power Level (kW/MTU)	Approximate Decay Time (days)	Propagation
11.0	5.9	365	Yes
19.2	5.9	365	Yes
90	5.9	365	Yes
90	4.0	730	No

Table 3.3 Summary of Radial Oxidation Propagation Results for a Cylindrical PWR Spent Fuel Rack with a 3 Inch Diameter Hole and Perfect Ventilation

High Power Level (kW/MTU)	Adjacent Power Level (kW/MTU)	Approximate Decay Time (days)	Propagation
90	11.0	365	No
90	19	180	Yes*

*Note that this is an unlikely situation in that the conditions imply a six month period between discharges.

Table 3.4 Summary of Radial Oxidation Propagation Results for a Cylindrical PWR Spent Fuel Rack with a 1.5 Inch Diameter Hole and Perfect Ventilation

High Power Level (kW/MTU)	Adjacent Power Level (kW/MTU)	Approximate Decay Time (days)	Propagation ?
90	11.0	365	Yes
90	5.9	730	Yes
90	3.0	1100	No
15	11.0	365	Yes
15	5.9	730	No

Table 3.5 Summary of Radial Oxidation Propagation Results for Various PWR Spent Fuel Racks with No Ventilation

Spent Fuel Rack	High Power Level (kW/MTU)	Adjacent Power Level (kW/MTU)	Propagation
Cylindrical with 1.5 inch hole	90	5.9	Yes
Cylindrical with 1.5 inch hole	90	3.0	No (O_2 depletion)
Cylindrical with 3 inch hole	90	5.9	No
Cylindrical with 3 inch hole	19.2	11.0	Yes
High Density with 10 inch hole	90	4.0	No (O_2 depletion)

Table 3.6 Comparison of SNL Small Scale Oxidation Tests to Calculations with CLAD

Test	Air Flow Rate (lpm)	Peak Temperatures Data (°C)	Peak Temperatures CLAD (°C) Mid	Peak Temperatures CLAD (°C) Top
4	12	1570	1900	1400
5	28.3	~1850	1960	1660
6	56.6	>2000*	2100	1800

*Thermocouple failure.

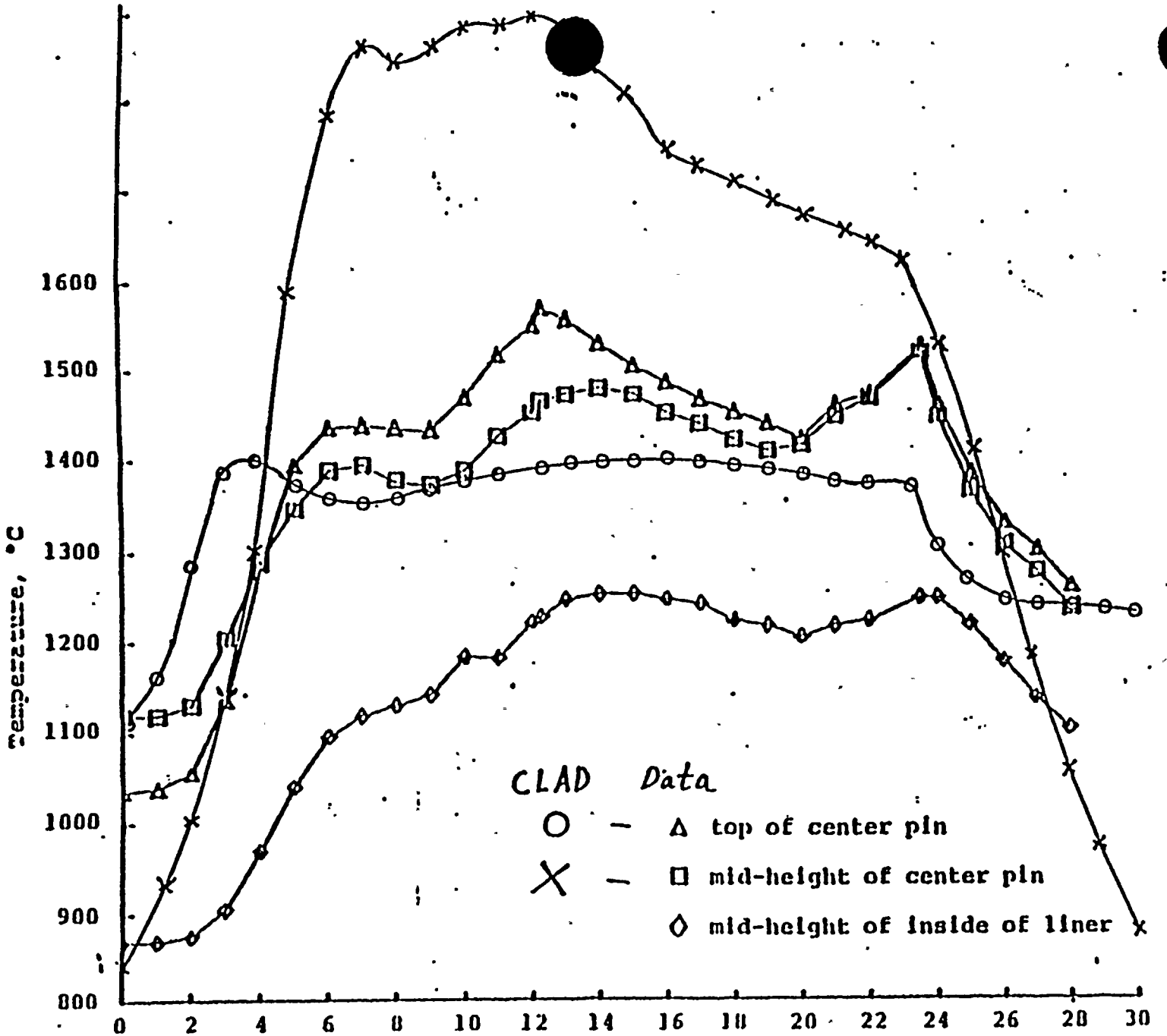

Figure 3.1 Comparison of CLAD to SNL data for Test 4.

4. CONSEQUENCE EVALUATION

A PWR and a BWR reactor were selected for risk evaluation based on a preliminary screening[1] of perceived vulnerability and the spent fuel pool inventory. The reactors selected were Ginna and Millstone 1. Both are older plants that were built before the current seismic design criteria were promulgated and have relatively large inventories of spent fuel.

4.1 Radionuclide Inventories

The radionuclide inventories for both the PWR and BWR pools were calculated using the ORIGEN2 Computer Code[2] for the actual operating and discharge histories for Ginna and Millstone 1. The ORIGEN2 program in use at BNL was verified by comparison with results obtained at ORNL for identical cases.[3]

A description of the assumptions and methods of analysis is given in Appendix A along with the detailed results for each species. The results for the risk significant species are summarized in Table 4.1 (Millstone 1) and Table 4.6 (Ginna).

For both plants, the noble gases and halogens in the spent fuel inventories are a small fraction of the inventory in an equilibrium core at shutdown except for freshly discharged fuel, but cesium and strontium are more than three times the equilibrium inventory (see Tables 4.1 and 4.6).

4.2 Release Estimates

The fission product release fractions have been calculated for two limiting cases in which a Zircaloy fire occurs: In Case 1, the clad combustion is assumed to propagate throughout the pool and the entire inventory is involved. In Case 2 only the most recently discharged fuel undergoes clad combustion.

The release calculations for Cases 1 and 2 make the assumption that if the spent fuel pool suffers a structural failure, coolant inventory will be totally drained, i.e., the leak rate will greatly exceed makeup capability

even if the coolant systems are still available. The probability of Zircaloy fire and fission product release has been determined from BNL calculations described in Section 3. In order for a cladding fire to occur the fuel must be recently discharged (about 10 to 150 days for a BWR and 30 to 250 days for a PWR). This leads to a conditional probability for a Zircaloy fire of .28 for a BWR and .40 for a PWR. If the discharged fuel is put into high density racks the critical cooling time is increased to one to three years and the conditional probability of a Zircaloy fire is increased to a virtual certainty.

A reevaluation of the cladding fire propagation estimates indicates that there is a substantial likelihood of propagation to other fuel bundles that have been discharged within the last one or two years. Subsequent propagation to low power bundles by thermal radiation is highly unlikely, but with a substantial amount of fuel and cladding debris on the pool floor, the coolability of even low power bundles is uncertain.

4.2.1 Estimated Releases for Self-Sustaining Cladding Oxidation Cases (Cases 1 and 2)

As discussed in Section 3.1 there are a broad range of spent fuel storage conditions for which self-sustaining oxidation of the cladding will occur if the water in the pool is lost. For Ginna with high density racks the conditional probability of a cladding "fire" is predicted to be nearly 100% while for Millstone 1 the probability is about 20%. If self-sustaining oxidation occurs the fuel rods are predicted to reach 1500 to 2100°C over a substantial portion of their length. At these temperatures, the release fraction is predicted to be substantial.

Rough estimates of the fractional release of various isotopes have been presented in an attachment to Ref. 4. Included in the estimates were noble gases (100%), halogens (100%), alkali metals (100%), tellurium (2 to 100%), barium (2%), strontium (0.2%) and ruthenium (0.002%).

Estimated release fractions of other isotopes are given in Table 4.2. These estimates are based on various considerations, including experimental

data (tellurium), location of the isotopes (whether in cladding as activation products or in fuel pellets as fission products), and melting/boiling points of the element or oxides of the element. Comments on the estimates listed in Table 4.2 follow:

Tellurium: The releases shown assume the lower limit of Ref. 4 based on the tellurium release model recently proposed by Lorenz, et al.[5] The low release value assumes that a fraction of the Zircaloy cladding relocates (melts and flows downward) before oxidation is complete.[5]

Alkali Earths: Because of the high boiling points of the oxides of Sr and Ba, it is estimated that only a very small fraction ($2x10^{-3}$) of these elements of fission product origen in the fuel pellets escape. It is estimated that 100% of the activation product Sr-89 and Y-91 contained in the Zircaloy cladding are released as aerosols.

Transition Elements: It is estimated that 100% of the transition element activation products contained in the cladding are levitated as aerosols of the oxides (smoke). Note that the small release fraction of Zr-95 (0.01) takes into account the large inventory of fission product Zr-95 trapped in the fuel pellets.

It is assumed that only 10% of the activation products in the assembly hardware escapes (see Table 4.2, Fe-55, Co-58, Co-60 and Y-91). The Co-60 fraction is corrected for its small content in the cladding.

Antimony: It is estimated that 100% of the SB-125 is roasted out of the fuel pellets, because of its high mobility.

Lathanides and Actinides: A negligible release of the oxides of the lathanides and actinides is estimated because of their chemical stability, low vapor pressures and ceramic characteristics.

Case 1: Case 1, the "worst" case, assumes an accident that results in a Zircaloy fire that propagates throughout the entire spent fuel inventory in the pool, and that the accident occurs 30 days after the reactor was shut down for discharge of the last fuel batch. The estimated releases of radionuclides are listed in Table 4.3. These were obtained by combining the "30-day" inventory given in Column 3 of Table 4.1 with the release fractions listed in Table 4.2.

Case 2: Case 2 assumes an accident that results in a Zircaloy fire that involves only the last fuel batch to be discharged, and that the accident occurs 90 days after the reactor was shut down for fuel discharge. The estimated releases of radionuclides are listed in Table 4.4. These were obtained by combining the inventory in the last fuel batch (data tabulated in Table A.6 of Appendix A) with the release fractions in Table 4.2.

4.2.2 Estimated Release for Low-Temperature Cladding Failure (Cases 3 and 4)

For a less severe accident in which fuel is exposed to air but does not reach temperatures at which a Zircaloy fire ignites, it is assumed that the cladding on many fuel rods will fail (i.e., develop leaks) resulting in a release limited to the noble gases and halogens. Two limiting cases have been considered:

Case 3: in which the entire pool is drained but the decay time since the last discharge is one year, and 50% of the fuel rods suffer clad rupture.

Case 4: in which the pool drains to a level that exposes the upper portion of the fuel assemblies, the decay time for the last discharged fuel batch is 30 days, no Zircaloy fire occurs but all of the fuel rods in the last discharged batch rupture.

The estimated releases for Cases 3 and 4 are given in Table 4.5.

4.3 Off-Site Radiological Consequences

4.3.1 Scenarios for Consequences Calculations

The off-site radiological consequences have been calculated using the CRAC2 computer code.[6] The scenario used in the CRAC2 calculations consisted of the following conditions:

- a generalized site surrounded by a constant population density of 100 persons per square mile;
- generalized meteorology (a uniform wind rose, average weather conditions); and
- the population in affected zones was relocated after 24 hours.

The radiological effects were calculated out to a distance of 50 to 500 miles.

CRAC2 calculations were made for a range of possible releases as described in Section 4.2. The consequences are summarized in Table 4.7.

4.3.2 Consequence Results

There are several unusual characteristics of a spent fuel accident that cause somewhat surprising results in the radiation exposure calculations. Specifically, the radiation exposure is insensitive to fairly large variations in the estimated release. This is due principally to the health physics assumptions within CRAC. For the long lived isotopes (predominantly cesium), the exposure is due mainly to exposure after the area is decontaminated and people return to their homes. The CRAC code assumes that decontamination will limit the exposure of each person to 25 rem. Thus, for this type of release the long term whole body dose is limited by the population in the affected sectors (about 0.8 million people in the 16 sectors for a 50 mile radius) to about 3×10^6 person-rem (only 3 of the 16 sectors are downwind).

The extreme cases (1A; immediately after refueling and 1B and 1C; with the total fuel pool inventory involved) result in much higher releases but no significant change in population dose.

A more sensitive indication of the consequences for a spent fuel accident is the interdiction area (the area with such a high level of radiation that it is assumed that it cannot ever be decontaminated). As indicated in Table 5 the worst spent fuel accident is calculated to result in an interdiction area of 224 sq. miles. This is about two orders of magnitude higher than the interdiction area computed for reactor core melt accidents (about 1 to 10 mi^2).

4.4 References for Section 4

1. BNL Memorandum, V.L. Sailor and K.R. Perkins to W.T. Pratt, "Study of Beyond Design Basis Accidents in Spent Fuel Pools," May 8 1985.

2. A.G. Croff, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials," Nuclear Technology, Vol. 62, pp. 335-352, September 1983.

3. Internal Memorandum, Brookhaven National Laboratory, from V.L. Sailor to R.A. Bari, "Comparison of BNL ORIGEN2 Calculations with ORNL," May 27, 1986.

4. Memorandum of J.T. Han to M. Silberberg, "Response to a NRR Request to Review SNL Studies Regarding Spent Fuel Heatup and Burning Following Loss of Water in Storage Pool," U.S. Nuclear Regulatory Commission, (May 21, 1984).

5. R.A. Lorenz, E.C. Beahm and R.P Wichner, "Review of Telliurium Release Rates from LWR Fuel Elements Under Accident Conditions," Proceedings of the International Meeting on Light Water Reactor Severe Accident Evaluation, August 28-September 1, 1983, pg. 4.4-1, American Nuclear Society Order 700085, ISBN 0-89448-1112-6.

6. L.T. Ritchie, J.D. Johnson and R.M. Blond, Calculations of Reactor Accident Consequences Version 2, CRAC2: Computer Code, User's Guide, prepared by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission, NUREG/CR-2326 (SAND81-1994), February 1983.

Table 4.1 Comparison of Radioactive Inventories of Equilibrium Core with Spent Fuel Assemblies for Selected Isotopes (Millstone 1)

Isotope	Equilibrium Core	Spent Fuel Pool[a] (time after last discharge) 30-days	90-days	1 year
	(Total Radioactivity, Curies)			
Co 58	8.81E+4	2.29E+4	1.26E+4	8.54E+2
Co 60	1.64E+5	3.72E+5	3.15E+5	2.85E+5
Kr 85	5.35E+5	1.41E+6	1.39E+6	1.33E+6
Rb 86	6.22E+4	1.01E+4	1.05E+3	3.84E-2
Sr 89	4.71E+7	8.39E+6	3.63E+6	8.33E+4
Sr 90	4.25E+6	1.42E+7	1.42E+7	1.39E+7
Y 90	4.37E+6	1.43E+7	1.42E+7	1.39E+7
Y 91	6.06E+7	1.18E+7	5.75E+6	2.21E+5
Zr 95	8.70E+7	1.94E+7	1.00E+7	5.10E+5
Nb 95	8.91E+7	2.54E+7	1.70E+7	1.11E+6
Mo 99	8.78E+7	1.49E+4	3.12E-3	neg.[b]
Tc 99m	7.69E+7	1.43E+4	3.01E-3	neg.[b]
Ru 103	7.23E+7	1.53E+7	5.21E+6	4.07E+4
Ru 106	2.48E+7	1.72E+7	1.53E+7	9.13E+6
Rh 106	2.63E+7	1.72E+7	1.53E+7	9.13E+6
Sb 125	9.07E+5	1.19E+6	1.14E+6	9.48E+5
Sb 127	4.97E+6	8.21E+3	1.39E-1	neg.[b]
Te 125m	1.93E+5	2.84E+5	2.76E+5	2.31E+5
Te 127	4.92E+6	2.21E+5	1.45E+5	2.52E+4
Te 127m	6.61E+5	2.18E+5	1.48E+5	2.57E+4
Te 129	1.49E+7	2.74E+5	7.79E+4	2.68E+2
Te 129m	2.24E+6	4.21E+5	1.20E+5	4.12E+2
Te 132	6.72E+7	3.74E+4	8.64E-2	neg.[b]
I 129	1.75E+0	7.15E+0	7.15E+0	7.15E+0
I 131	4.74E+7	1.22E+6	6.35E+3	neg.[b]
I 132	6.83E+7	3.85E+4	8.90E-2	neg.[b]
Xe 133	9.72E+7	7.29E+5	2.30E+2	neg.[b]
Cs 134	6.10E+6	7.90E+6	7.47E+6	5.80E+6
Cs 136	2.10E+6	2.05E+5	8.13E+3	3.91E-3
Cs 137	5.84E+6	2.02E+7	2.01E+7	1.97E+7
Ba 137m	5.53E+6	1.91E+7	1.90E+7	1.87E+7
Ba 140	8.36E+7	5.19E+6	1.90E+5	6.41E-2
La 140	8.54E+7	5.97E+6	2.19E+5	7.37E-2
Ce 141	7.94E+7	1.32E+7	3.61E+6	1.03E+4
Ce 144	6.05E+7	2.64E+7	2.27E+7	1.16E+7
Pr 143	7.37E+7	5.44E+6	2.41E+5	1.90E-1
Pr 144	6.08E+7	2.64E+7	2.27E+7	1.16E+7
Nd 147	3.16E+7	1.54E+6	3.36E+4	1.10E-3
Sm 151	2.44E+4	8.22E+4	8.21E+4	8.16E+4
Eu 154	4.61E+5	1.34E+6	1.32E+6	1.25E+6
Eu 156	5.61E+6	8.26E+5	5.10E+4	1.80E-1
Np 239	9.98E+8	5.59E+4	2.88E+3	2.88E+3
Pu 238	9.33E+4	4.51E+5	4.53E+5	4.54E+5
Pu 239	2.49E+4	8.89E+4	8.89E+4	8.89E+4
Pu 240	3.14E+4	1.30E+5	1.30E+5	1.30E+5
Pu 241	7.19E+6	2.29E+7	2.27E+7	2.19E+7
Am 241	8.86E+3	2.88E+5	2.94E+5	3.21E+5
Cm 242	2.09E+6	1.45E+6	1.12E+6	3.50E+5
Cm 244	6.72E+4	2.27E+5	2.25E+5	2.19E+5

[a]Spent fuel pool inventory includes discharges from 11 refuelings covering the period from August 1972 through the projected refueling of April 1987.

[b]neg. = less than 10^{-3} Curies.

Table 4.2 Estimated Radionuclide Release Fraction During a Spent Fuel Pool Accident Resulting in Complete Destruction of Cladding (Cases 1 and 2)

Chemical Family	Element or Isotope	Release Fraction[a] Value Used	Release Fraction[a] Uncertainty Range
Noble gases	Kr, Xe	1.00	0
Halogens	I-129, I-131	1.00	0.5-1.0
Alkali Metals	Cs, (Ba-137m) Rb	1.00	0.5-1.0
Chalcogens	Te, (I-132)	0.02	
Alkali Earths	Sr, (Y-90), Ba (in fuel)	2×10^{-3}	10^{-4}-10^{-2}
	Sr, Y-91 (in clad)	1.00	0.5-1.0
Transition Elements	Co-58 (assembly hardware)	0.10	0.1-1.0
	Co-60 (assembly hardware)[b]	0.12	0.1-1.0
	Y-91 (assembly hardware)	0.10	0.1-1.0
	Nb-95, Zr-95 (in fuel)	0.01	10^{-3}-10^{-1}
	Nb-95, Zr-95 (in clad)	1.00	0.5-1.0
Miscellaneous	Mo-99	1×10^{-6}	10^{-8}-10^{-5}
	Ru-106	2×10^{-5}	10^{-6}-10^{-4}
	Sb-125	1.00	0.5-1.0
Lanthanides	La, Ce, Pr, Nd, Sm, Eu	1×10^{-6}	10^{-8}-10^{-5}
Transuranics	Np, Pu, Am, Cm	1×10^{-6}	10^{-8}-10^{-5}

[a]Release fractions of several daughter isotopes are determined by their precursors, e.g., Y-90 by Sr-90, Tc-99m by Mo-99, Rh-106 by Ru-106, I-132 by Te-132, Ba-137m by Cs-137, and La-140 by Ba-140.

[b]Release fraction adjusted to account for a 100% release of the small amount of Co-60 contained in the Zircaloy cladding.

Table 4.3 Estimated Releases of Radionuclides for Case 1 in Which a Zircaloy Fire Propagates Throughout the Entire Pool Inventory (Worst Case)

Isotope	Time after Last Discharge		
	30-days	90-days	1 year
	(Radioactivity, Curies)		
Co 58	2.74E+3	1.51E+3	1.02E+2
Co 60	4.46E+4	3.78E+4	3.42E+4
Kr 85	1.41E+6	1.39E+6	1.33E+6
Rb 86	1.01E+4	1.05E+5	3.84E-2
Sr 89	1.68E+4	7.26E+3	1.67E+2
Sr 90	2.84E+4	2.84E+4	2.78E+4
Y 90	2.84E+4	2.84E+4	2.78E+4
Y 91	1.18E+6	5.75E+5	2.21E+4
Zr 95	1.63E+6	8.39E+5	4.26E+4
Nb 95	2.13E+6	1.42E+6	9.27E+4
Mo 99	1.49E-2	neg.[a]	neg.[a]
Tc 99m	1.43E-2	neg.[a]	neg.[a]
Ru 103	3.06E+2	1.04E+2	8.14E-1
Ru 106	3.44E+2	3.06E+2	1.83E+2
Rh 106	3.44E+2	3.06E+2	1.83E+2
Sb 125	1.19E+6	1.14E+6	9.48E+5
Sb 127	8.21E+3	1.39E-1	neg.[a]
Te 125m	5.68E+3	5.52E+3	4.62E+3
Te 127	4.42E+3	2.90E+3	5.04E+2
Te 127m	4.36E+3	2.96E+3	5.14E+2
Te 129	5.48E+3	1.56E+3	5.36E+0
Te 129m	8.42E+3	2.40E+3	8.24E+0
Te 132	7.48E+2	1.72E-3	neg.[a]
I 129	7.15E+0	7.15E+0	7.15E+0
I 131	1.22E+5	6.35E+3	neg.[a]
I 132	7.70E+2	1.78E-3	neg.[a]
Xe 133	7.29E+5	2.30E+2	neg.[a]
Cs 134	7.90E+6	7.47E+6	5.80E+6
Cs 136	2.05E+5	8.13E+3	3.91E-3
Cs 137	2.02E+7	2.01E+7	1.97E+7
Ba 137m	1.91E+7	1.90E+7	1.87E+7
Ba 140	1.04E+4	3.80E+2	neg.[a]
La 140	1.19E+4	4.38E+2	neg.[a]
Ce 141	1.32E+1	3.61E+0	1.03E-2
Ce 144	2.64E+1	2.27E+1	1.16E+1
Pr 143	5.44E+0	2.41E-1	neg.[a]
Pr 144	2.64E+1	2.27E+1	1.16E+1
Nd 147	1.54E+0	3.36E-2	neg.[a]
Sm 151	8.22E-2	8.21E-2	8.16E-2
Eu 154	1.34E+0	1.32E+0	1.25E+0
Eu 156	8.26E-1	5.10E-2	neg.[a]
Np 239	5.59E-2	2.88E-3	2.88E-3
Pu 238	4.51E-1	4.53E-1	4.54E-1
Pu 239	8.89E-2	8.89E-2	8.89E-2
Pu 240	1.30E-1	1.30E-1	1.30E-1
Pu 241	2.29E+1	2.27E+1	2.19E+1
Am 241	2.28E-1	2.94E-1	3.21E-1
Cm 242	1.45E+0	1.12E+0	3.50E-1
Cm 244	2.27E-1	2.25E-1	2.19E-1

[a]neg. = less than 10^{-3} Curies.

Table 4.4 Estimated Releases of Radionuclides for Case 2 in Which Only the Last Discharged Fuel Batch Suffers a Zircaloy Fire

Isotope	Time after Last Discharge		
	30-days	90-days	1 year
	(Radioactivity, Curies)		
Co 58	2.28E+3	1.26E+3	8.49E+1
Co 60	9.17E+3	8.68E+3	8.12E+3
Kr 85	2.39E+5	2.36E+5	2.25E+5
Rb 86	1.01E+4	1.05E+3	3.84E-2
Sr 89	1.79E+4	7.75E+3	1.78E+2
Sr 90	3.84E+3	3.82E+3	3.78E+3
Y 90	3.86E+3	3.84E+3	3.78E+3
Y 91	2.66E+4	1.30E+4	4.99E+2
Zr 95	1.62E+6	8.37E+5	4.25E+4
Nb 95	2.11E+6	1.41E+6	9.24E+4
Mo 99	1.49E-2	neg.[a]	neg.[a]
Tc 99m	1.43E-2	neg.[a]	neg.[a]
Ru 103	3.06E+2	1.04E+2	8.14E-1
Ru 106	2.24E+2	1.99E+2	1.19E+2
Rh 106	2.24E+2	1.99E+2	1.19E+2
Sb 125	4.17E+5	4.00E+5	3.31E+5
Sb 127	8.21E+3	1.39E-1	neg.[a]
Te 125m	1.88E+3	1.88E+3	1.61E+3
Te 127	4.28E+3	2.80E+3	4.86E+2
Te 127m	4.20E+3	2.86E+3	4.96E+2
Te 129	5.48E+3	1.56E+3	5.36E+0
Te 129m	8.42E+3	2.40E+3	8.24E+0
Te 132	7.48E+2	1.73E-3	neg.[a]
I 129	8.84E-1	8.86E-1	8.86E-1
I 131	1.22E+6	6.35E+3	neg.[a]
I 132	7.70E+2	1.78E-3	neg.[a]
Xe 133	7.29E+5	2.30E+2	neg.
Cs 134	3.53E+6	3.34E+6	2.59E+6
Cs 136	2.05E+5	8.13E+3	3.91E-3
Cs 137	2.83E+6	2.81E+6	2.77E+6
Ba 137m	2.67E+6	2.66E+6	2.62E+6
Ba 140	1.04E+5	3.80E+3	1.28E-3
La 140	1.19E+4	4.38E+2	neg.[a]
Ce 141	1.32E+1	3.61E+0	1.03E-2
Ce 144	1.91E+1	1.65E+1	8.43E+0
Pr 143	5.44E+0	2.41E-1	neg.[a]
Pr 144	1.91E+1	1.65E+1	8.43E+0
Nd 147	1.54E+0	3.36E-2	neg.[a]
Sm 151	9.31E-3	9.30E-3	9.25E-3
Eu 154	2.89E-1	2.85E-1	2.69E-1
Eu 156	8.37E-1	5.82E-2	neg.[a]
Np 239	5.36E-2	neg.[a]	neg.[a]
Pu 238	6.73E-2	6.87E-2	7.18E-2
Pu 239	9.28E-3	9.28E-3	9.28E-3
Pu 240	1.55E-2	1.55E-2	1.55E-2
Pu 241	3.73E+0	3.70E+0	3.56E+0
Am 241	6.01E-3	7.00E-3	1.14E-2
Cm 242	1.31E+0	1.01E+0	3.16E-1
Cm 244	5.88E-2	5.84E-2	5.68E-2

[a] neg. = less than 10^{-3} Curies.

Table 4.5 Estimated Releases of Radionuclides for Cases 3 and 4 in Which Low-Temperature Cladding Failures Occur

Isotope	Case 3[a]	Case 4[b]
	(Radioactivity, Curies)	
Kr 85	6.65E+5	2.39E+5
I 129	3.58E+0	8.84E-1
I 131	neg.[c]	1.22E+6
I 132	neg.[c]	7.70E+2
Xe 133	neg.[c]	7.29E+5

[a]Case 3 assumes:

1. last fuel discharge has decayed for 1 year.
2. no Zircaloy fire occurs.
3. 50% of the fuel rods develop leaks.
4. 100% release of noble gases and halogens from leaking fuel rods.

[b]Case 4 assumes:

1. last fuel batch discharged has decayed for 30 days.
2. no Zircaloy fire occurs.
3. 100% of fuel rods in last discharge devlop leaks.
4. 100% release of noble gases and halogens from leaking fuel rods.

[c]neg. = less than 10^{-3} Curies.

Table 4.6 Comparison of Radioactive Inventories of Equilibrium Core with Spent Fuel Assemblies for Selected Isotopes (Ginna).

Isotope	Equilibrium Core	Spent Fuel Pool[a] (time after last discharge) 30-days	90-days	1 year
	(Total Radioactivity, Curies)			
Co 58	3.57E+5	5.93E+4	3.26E+4	2.21E+3
Co 60	3.20E+5	5.97E+5	5.84E+5	5.29E+5
Kr 85	3.73E+5	9.84E+5	9.74E+5	9.27E+5
Rb 86	6.53E+5	7.22E+3	7.48E+2	2.74E-2
Sr 89	3.55E+7	3.53E+6	1.53E+6	3.50E+4
Sr 90	2.95E+6	1.02E+7	1.01E+7	9.95E+6
Y 90	3.15E+6	1.02E+7	1.01E+7	9.95E+6
Y 91	4.57E+7	5.11E+6	2.48E+6	9.54E+4
Zr 95	6.41E+7	8.64E+6	4.46E+6	2.27E+5
Nb 95	6.34E+7	1.12E+7	7.51E+6	4.93E+5
Mo 99	6.83E+7	7.03E+3	1.48E-3	neg.[b]
Tc 99m	5.89E+7	6.77E+3	1.42E-3	neg.[b]
Ru 103	5.85E+7	7.86E+6	2.88E+6	2.09E+4
Ru 106	1.95E+7	1.09E+7	9.71E+6	5.78E+6
Rh 106	2.15E+7	1.09E+7	9.71E+6	5.78E+6
Sb 125	6.04E+5	7.11E+5	6.82E+5	5.65E+5
Sb 127	4.12E+6	4.33E+3	7.35E-2	neg.[b]
Te 125m	1.27E+5	1.70E+5	1.65E+5	1.37E+5
Te 127	4.05E+6	1.19E+5	7.79E+4	1.36E+4
Te 127m	5.19E+5	1.17E+5	7.95E+4	1.38E+4
Te 129	1.21E+7	1.38E+5	3.93E+4	1.35E+2
Te 129m	1.80E+6	2.12E+5	6.03E+4	2.07E+2
Te 132	5.33E+7	1.83E+4	4.23E-2	neg.[b]
I 129	1.27E+0	5.32E+0	5.32E+0	5.32E+0
I 131	3.76E+7	6.00E+5	3.12E+3	neg.[b]
I 132	5.42E+7	1.89E+4	4.36E-2	neg.[b]
Xe 133	7.64E+7	3.52E+5	1.11E+2	neg.[b]
Cs 134	5.82E+6	6.35E+6	6.00E+6	4.66E+6
Cs 136	1.87E+6	1.26E+5	4.99E+3	2.40E-3
Cs 137	4.21E+6	1.48E+7	1.47E+7	1.44E+7
Ba 137m	4.00E+6	1.40E+7	1.39E+7	1.37E+7
Ba 140	6.55E+7	2.47E+6	9.07E+4	3.05E-2
La 140	6.74E+7	2.85E+6	1.04E+5	3.51E-2
Ce 141	6.28E+7	6.34E+6	1.72E+6	4.91E+3
Ce 144	4.24E+7	1.38E+7	1.19E+7	6.09E+6
Pr 143	5.71E+7	2.54E+6	1.12E+5	8.86E-2
Pr 144	4.27E+7	1.38E+7	1.19E+7	6.09E+6
Nd 147	2.48E+7	7.42E+5	1.62E+4	neg.[b]
Sm 151	1.42E+4	5.14E+4	5.13E+4	5.10E+4
Eu 154	4.09E+5	1.09E+6	1.07E+6	1.01E+6
Eu 156	7.22E+6	7.58E+5	4.68E+4	1.66E-1
Np 239	7.81E+8	3.02E+4	3.26E+3	3.26E+3
Pu 238	1.01E+5	4.46E+5	4.46E+5	4.46E+5
Pu 239	1.35E+4	5.25E+4	5.25E+4	5.25E+4
Pu 240	2.02E+4	8.60E+4	8.60E+4	8.61E+4
Pu 241	4.85E+6	1.52E+7	1.51E+7	1.46E+7
Am 241	4.99E+3	2.10E+5	2.14E+5	2.32E+5
Cm 242	1.91E+6	9.33E+5	7.20E+5	2.25E+5
Cm 244	1.25E+5	3.59E+5	3.56E+5	3.46E+5

[a]Spent fuel pool inventory includes discharges from 15 refuelings covering the period from April 1983 through the projected refueling of April 1987.

[b]neg. = less than 10^{-3} Curies.

Table 4.7 CRAC2 Results for Various Releases Corresponding to Postulated Spent Fuel Pool Accidents with Total Loss of Pool Water

	Case Description	Whole Body Dose (Man-rem)	Interdiction Area (sq. miles)
1A.	Total inventory 30 days after discharge 50 mile radial zone	2.6×10^6	224
1B.	Total inventory 90 days after discharge 50 mile radial zone	2.6×10^6	215
1C.*	Total inventory 30days after discharge 500 mile radial zone	7.1×10^7	224
2.	Last fuel discharge 90 days after discharge 50 mile radial zone	2.3×10^6	44

*Note that the consequence calculations in NUREG-1150 are based on a 50 mile radial zone. Case 1C is given as a sensitivity result.

5. RISK PROFILE

The likelihood and consequences of various spent fuel pool accidents has been estimated in the previous sections. The risk is summarized in Table 5.1. As previously mentioned, the exposure results are tied to the health physics assumptions regarding decontamination and maximum allowable exposure. Thus the land interdiction area is included in Table 5.1 as a more meaningful representation of severity. The uncertainty in each of these risk indices is estimated to be an order of magnitude in either direction and is due principally to uncertainty in the fragility of the pools and uncertainty in the seismic hazard.

Note that the risk results are calculated for two surrogate plants and may not be applicable to generic pool types.

5.1 Failure Frequency Estimates

5.1.1 Spent Fuel Pool Failure Probability

The likelihood of the various postulated spent fuel pool accidents was developed in Section 2 and summarized in Table 2.9. The probability is similar to the frequency of dominant core melt sequences for many PRA's. The major contributors are:

1. Cask drop accidents,
2. Seismic induced pool failure,
3. Loss of pool cooling, and
4. Pneumatic seal failure.

Note that all of these potential accidents are plant specific and their frequency will vary widely from plant to plant. In particular, BWR's do not have pneumatic seals so their failure frequency is zero.

5.1.2 Spent Fuel Failure Likelihood

Previous investigations[1,2] of spent fuel behavior after a loss of pool integrity accident focused on the conditions necessary to initiate cladding "fires" after a spent fuel pool has drained. The present project has reevaluated these conditions using the SFUEL code[2] developed by SNL. The likelihood of such cladding fires has been assessed in Section 3. For a PWR with high density storage racks, the conditional probability of a clad fire was found to be 1.0 while for a BWR with low density storage racks the probability of a clad fire was found to be 0.08.

5.2 Conclusions Regarding Risk

The overall risk due to beyond design basis accidents in spent fuel pools for the PWR surrogate plant is about 130 person-rem/Ry and about 12 person-rem/Ry for the BWR surrogate. These estimates are comparable to present estimates[3] for dominant core melt accidents and appear to warrant further attention on this basis alone. However, the unique character of such an accident (substantial releases of long lived isotopes) makes it difficult to compare to reactor core melt accidents. The exposure calculations are driven by assumptions in the CRAC modeling and the results are not sensitive to the severity of the accident. In terms of interdiction area this type of accident has the potential to be much worse than a reactor core melt accident.

The uncertainty in risk in terms of person-rem/Ry is driven principally by the uncertainty in the likelihood of complete draining of the spent fuel pool which is estimated to be at least an order of magnitude in either direction.

5.3 References for Section 5

1. A.S. Benjamin, D.J. McCloskey, D.A. Powers, S.A. Dupree, "Spent Fuel Heat-up Following Loss of Water During Storage," NUREG/CR-0649, March 1979.

2. N.A. Pisano, F. Best, A.S. Benjamin, K.T. Stalker, "The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool," Draft Report, January 1984.

3. "Evaluation of Severe Accident Risks and the Potential for Risk Reduction," NUREG-1150 (To be published).

Table 5.1 Estimated Risk for the Two Surrogate Spent Fuel Pools from the Two Dominant Contributors

Accident Initiator	Spent Fuel Pool Fire Probability/Ry	Health Risk (Man-rem/Ry)	Interdiction Risk (Sq. Mi./Ry)
Seismic induced PWR pool failure	1.6×10^{-5}	37	8.4×10^{-4}
Seismic induced BWR pool failure	1.8×10^{-6}	4	7.6×10^{-5}
Cask drop* induced PWR pool failure	3.1×10^{-5}	71	.001
Cask drop* induced BWR pool failure	2.5×10^{-6}	6	1.1×10^{-4}

*After removal of accumulated inventory resumes. (Note that many new plants have pool configurations and administrative procedures which would preclude this failure mode.)

6. CONSIDERATION OF RISK REDUCTION MEASURES

Due to diversity in the nature of initiating events for beyond design basis accidents in spent fuel pools, there appear to be several possible ways to reduce the risks. It must be emphasized that each of the contributors to risk are plant specific and one or more of the risk significant sequences identified in Section 5 may not be important at other plant sites. The following sections discuss the advantage and disadvantages of a number of proposed risk reduction strategies. A cost benefit analysis has not been performed but the estimated risk appears to be large enough to justify further investigation of risk reduction measures.

6.1 Risk Prevention

1. Reduction of Stored Radioactive Inventory - Most of the consequences of a release of radioactivity from a catastropic pool accident is associated with the large inventory of isotopes of intermediate half-lives, e.g., Cs-137, Sr-90. The potential release increases approximately in proportion to the number of fuel assemblies in the storage inventory. One obvious measure for risk reduction is to transfer part of the inventory to alternative storage locations (e.g., see Ref. 1).

2. Air Circulation - The one universal prevention measure is to promote air cooling in the event of loss of water cooling of the spent fuel. The new high density fuel storage racks restrict air flow and make even old spent fuel (one to two years) susceptible to heat-up and self-sustaining oxidation. The older style fuel baskets with large inlet holes (3 inch diameter or more per assembly) allow much freer air circulation. If all recently discharged fuel (less than two years) is kept in low density fuel baskets and they are separated from the wall and the older fuel by a one foot gap then the likelihood of self-sustaining oxidation would be reduced by a factor of 5 or more compared to the high density storage configuration.

3. Additional Cooling Systems - Although loss-of-pool cooling appears to be risk significant, an additional cooling system is unlikely to be cost beneficial (unless the cooling system was substantially more unreliable than the two surrogate systems). An additional cooling system would not affect the risk from pool failure events (seismic or cask drop accidents). Thus the net risk reduction would be minimal unless loss-of-cooling were the dominant event.

4. Improved Procedures and Equipment - The likelihood of cask drop accidents can be reduced by improving procedures, administrative controls and/or installing more reliable equipment. However, none of these improvements would reduce the risk from the other dominant sequences. Thus the net risk reduction would be difficult to quantify on a plant specific basis. It would appear to be useful to conduct a complete risk evaluation before spent fuel shipment is begun at each site. A key piece of such an evaluation would be a structural analysis of the pool response to the loading from a dropped cask.

6.2 Accident Mitigation

1. Post-Accident Spray - Water spray has the potential to terminate the progression of a spent fuel pool accident whether or not the pool is intact. However, large quantities of water must be available (it would be necessary to continue spraying until the pool could be repaired and reflooded) and the equipment would have to be seismically qualified to a higher g level than the pool structure (in order for the sprays to have a high likelihood of surviving). Some pools may have fire sprays available in the spent fuel pool building. For those plants without sprays available, it seems unlikely that the expense of a new safety grade spray system could be justified considering the large uncertainty in the risk. Temporary fire hoses were suggested by Benjamin et al.,[2] but the radiation levels would make such ad hoc measures extremely difficult. Furthermore, if the spray is not initiated before the rods reach 900 C or there is insufficient flow, the water may aggravate the reaction by providing additional oxidation. (The steam/Zircaloy reaction is also highly exothermic.)

2. Filtering - For those plants with a standby gas treatment system available, operation of the system has the potential to substantially reduce the fission product release from the building. However, the high temperatures and large aerosol production rate would tend to rapidly degrade the effectiveness of the system. The performance of such a filtering system would be difficult to characterize under fuel pool accident conditions. It is unlikely to be cost effective to install a new system large enough to handle the worst case spent fuel pool accident scenarios.

6.3 Conclusions Regarding Preventive and Mitigative Measures

For those plants which have a similar spent fuel pool risk potential to the two surrogate plants, the one preventive measure which appears to have a substantial effect on risk (a risk reduction of 5 or more) is to maintain recently discharged fuel in low density storage racks that are isolated from the rest of the fuel racks by a foot or more of space (to provide free air circulation). However, there may be plant specific features which make a substantial difference in the order of the dominant contributors to risk. Therefore plant specific risk evaluations should be performed before any changes are implemented at a given plant.

6.4 References for Section 6

1. D.D. Orvis, C. Johnson, and R. Jones, "Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment," prepared for the Electric Power Research Institute by the NUS Corporation, EPRI NP-3365, February 1984.

2. A.S. Benjamin, D.J. McCloskey, D.A. Powers, S.A. Dupree, "Spent Fuel Heat-up Following Loss of Water During Storage," NUREG/CR-0649, March 1979.

APPENDIX A

RADIOACTIVE INVENTORIES

A.1 INTRODUCTION

Two older-vintage plants, a BWR and a PWR, were selected to serve as surrogates for estimating the risks associated with "Beyond Design Basis Accidents in Spent Fuel Pools." The purpose of this appendix is to describe the methods used to simulate the operating history of the two plants and to summarize the calculated radioactive inventories contained in the fuel assemblies stored in the spent fuel basins. The surrogate plants were Millstone-1 (BWR) and Ginna (PWR).

A.2 SIMULATION OF OPERATING HISTORIES

A.2.1 Thermal Energy Production vs Time

The operating history of each surrogate plant was reconstructed from several sources. The early history, prior to December 1, 1975 was reconstructed from monthly summaries contained in Refs. 1-3. Data for the period December 1, 1975 through April 30, 1986 were taken from Ref. 4. Data from May 1, 1986 to April 1, 1987 were extrapolated, based on recent average capacity factors and scheduled shutdowns.

During each operating cycle (the period between successive refuelings), the average thermal power was calculated from the total thermal energy produced during the cycle. No attempt was made to model variations in power levels during an operating period. (Fluctuations in the monthly energy production are illustrated in Fig. A.1.)

A.2.2 Fuel Burnup Calculations

The number of fuel assemblies discharged at each refueling and their specific burnup was obtained from a data base maintained by R.A. Libby of Pacific Northwest Laboratories (PNL) for the U.S. Department of Energy.[5] It should be

noted that the inventory of spent fuel assemblies stored in the spent fuel basins at various points in time listed in the Libby data base differ from the data listed in Ref. 4. It is apparent from the operating histories that the data in the earlier volumes of Ref. 4 are less accurate.

In general, the burnups listed in the Libby data base differ by a few percent from the burnups calculated by the methods described in the following paragraphs. These discrepancies do not have significant effects on the overall inventories of radionuclides, but only on the distribution of the inventories among the older fuel batches.

In order to model the burnup of the various discharged batches of spent fuel, the following method was used. It was assumed that all fuel assemblies in the core during a given operating cycle provided the average specific power, i.e.,

$$(MW_{th}/MT)_i = (MW_{th}D)_i/D_i(MT)_{core} ,$$

where for operating cycle, i, MW_{th}/MT is the average specific power per metric tonne of initial heavy metal, $(MW_{th}D)_i$ is the total thermal energy produced in D_i days of the cycle, i, and MT_{core} is the metric tonnes of initial heavy metal in the core.

The average specific burnup for each fuel batch, j, at discharge was calculated from the formula,

$$(MW_{th}D/MT)_j = \sum_i (MW_{th}/MT)_i D_i ,$$

where $\sum_i$ is the summation over the several operating cycles, i, that the fuel was in the reactor. (As noted below, ORIGEN2 also calculates the specific burnup which provides a check on internal consistency of the data).

The total burnup in the discharged fuel plus the burnup of assemblies remaining in the core at the time of the April 1, 1987 refueling equaled the

total thermal energy production over the preceding history of the plant (e.g., see Table A.4).

A.2.3 Calculation of Radioactive Inventories

The average radionuclide content in each metric tonne of discharged fuel was calculated using the ORIGEN2 Computer Code.[6] The code treats the reactor core as a homogeneous body operating at an average specific power. Account is taken of radionuclide decay during and following irradiation, decay chains, and successive neutron captures.

The BNL version of ORIGEN2 was benchmarked against the version in use at Oak Ridge National Laboratory by calculating an identical case, which yielded identical results.[7]

The results obtained from an ORIGEN2 calculation are slightly sensitive to the size of the time steps used in the irradiation calculation. Several preliminary calculations were made to select an appropriate set of time steps for which the sensitivity was negligible. (Shorter time steps give higher precision results, but at the expense of increased computer time. The criterion adopted was that the time-step sensitivity be less than 0.1% in the calculated concentration of several key nuclides.)

In a mature operating nuclear power plant fuel management strategies are complicated (e.g., see Ref. 8). Most fuel assemblies remain in the core for several operating cycles and are often shifted in location during refueling so as to optimize burnup. Also, U-235 enrichment is varied. ORIGEN2 as used at BNL did not take account of such detail, nor of the axial and radial distribution of the power density. Thus, the radioactivity calculated for a particular assembly would not correspond exactly to an actual assembly. Nevertheless, the total calculated radioactivity in a discharged batch should be identical to total in a real batch (in so far as the precision of ORIGEN2 allows).

The calculations do take account of the irradiation times in each operating cycle and the decay that occurs during shutdowns for refueling or prolonged shutdowns for maintenance and repair.

As used at BNL, the input for each irradiation cycle is the average specific power and the length of the cycle. ORIGEN2 calculates the total average burnup of each fuel batch over the irradiation cycles during which it was in the core. This calculated burnup was cross checked against "hand" calculations for each batch, the "hand" calculations being based on the operating history (see Section A.2.2).

The input for ORIGEN2 requires the specification of the elements contained in the fuel including trace impurities, the U-235 enrichment and the composition and amount of alloys used in the fuel cladding and assembly hardware. For each plant, BWR and PWR, only a single fuel and assembly composition was modeled which is typical of fuel of recent vintage for the respective reactors. Data for the fuel models were taken from Reference 9.

The output of ORIGEN2 includes isotopic concentrations (of stable as well as radioactive isotopes), activity of radionuclides, and thermal power production of each radionuclide. These are given at specified decay times for activation products (in cladding, hardware and trace elements in the fuel pellets), fission products and actinides.

The BNL calculations were made for each fuel batch from the date of the end of irradiation to the projected dates of May 1, 1987, July 1, 1987, October 1, 1987 and April 1, 1988.

A.3 DATA FOR MILLSTONE 1

A.3.1 Reactor and Fuel Cycle Parameters

Table A.1 summarizes several of the major reactor characteristics and fuel cycle parameters for Millstone 1.

A.3.2 History of Operations

Several milestones in the operation of Millstone-1 are summarized in Table A.2. Monthly gross thermal energy production from 1976 through 1984 is plotted in Fig. A.1. During the first 10 years of operation the plant

experienced two prolonged outages, i.e. Sept. 1972 to March 1973 (198 days) and October 1980 to June 1981 (254 days). Otherwise the refueling/maintenance outages have ranged from 35 to 76 days in duration averaging about 57 days.

A more detailed narrative of the plant operating history from 1970 through 1981 appears in Ref. 10, Appendix F, pp. F-31 through F-70. The only unusual experience with fuel cladding failures that has been noted occurred in 1974 when some 25 assemblies were found to have leaking fuel elements which forced a temporary power derating to stay within off-gas release limits.[11] Since mid-1981, the plant has operated with nearly 100% unit service factor except for scheduled refueling outages.[4]

There have been 10 refueling campaigns since beginning of commercial operations on March 1, 1971 (see Table A.3). The next scheduled refueling will be about April 1987. During the first 10 years, refueling occurred at somewhat irregular intervals, being dictated by unscheduled forced outages. Since 1981, refueling has been scheduled for approximately 18 month intervals, occurring in April or September. During the lifetime of the plant the average fuel burnup has generally increased from about 20,000 MWD/MT in 1972 to about 28,000 MWD/MT at present.

A.3.3 BWR Fuel Assembly Model Used in ORIGEN2 Calculations

A nominal BWR fuel element has been modeled, based on data presented in Ref. 9. This is an 8x8 element assembly of 2.75% U-235 enrichment, containing 1.5873 kg of gadolinium burnable poison per metric tonne of uranium. The fuel cladding is Zircaloy-2. Other alloys present in the fuel assembly hardware include Zircaloy-4, Inconel X-750, SS302 and SS304. The alloy contents of the assembly hardware are included with weighting factors to take account of the axial variation of neutron flux which results in lower neutron activation at the ends of the assemblies. In addition to the fuel, the cladding and the assembly hardware, an allowance was made for the presence of "crud" composed of Fe, Co, and Ni on the outer surfaces of the cladding and assembly hardware.

A.3.4 Calculated Radioactive Inventories

The calculated inventories of selected radionuclides* are listed in Table A.5 for the reactor core at the end of operating cycle number 11 projected to be on April 1, 1987. Also listed are the inventories in the spent fuel basin on May 1 and July 1, 1987 and April 1, 1988 assuming that 167 assemblies will be discharged in the April 1987 refueling.

It should be noted that many of the isotopes that are of considerable importance in a core melt accident are those of short half-lives which are no longer present in the spent fuel after a few days of decay, e.g., Rb-91, Rb-93, Sr-93, Sr-95, Y-94, Y-95, Tc-104, I-134, I-135, I-136, Cs-138, Cs-140. On the other hand, the spent fuel inventory contains much larger quantities of several long-lived isotopes than does the equilibrium core. Noteworthy among these are H-3, C-14, Sr-90 (Y-90), I-129, Cs-137, Ba-137m, Eu-154, Pu-239, Pu-240, Pu-241, Am-241, and Cm-244.

Table A.6 gives a comparison of the radionuclide inventories in the last fuel batch to be discharged with the summation of the inventories contained in the ten batches discharged in the period from 1972 through 1985.

A.3.5 Decay Heat

Table A.7 summarizes the decay thermal production in the various discharged batches. The data shown is for the whole batch, i.e., the specific thermal power (kilowatts per metric tonne) multiplied by the metric tonnes in the batch.

Table A.8 summarizes the fraction of the decay heat contributed by various isotopes. The main contributors change with decay time, e.g., in the oldest fuel (batches 1, 2, etc.) the largest contributors are Y-90 and Ba-137m, whereas the last discharged batch 11 is dominated by Cs-134, Rh-106, and Pr-144. The actinides are relatively small contributors.

*The selection of radionuclides was based on several criteria including potential for biological concern, thermal power, and total curies of activity.

A.4 DATA FOR GINNA

A.4.1 Reactor and Fuel Cycle Parameters

Table A.9 summarizes several of the major reactor characteristics and fuel cycle parameters for Ginna.

A.4.2 History of Operations

Several milestones in the operation of Ginna are summarized in Table A.10. A narrative of the operating history from 1969 through 1979 can be found in Ref. 12, Appendix F.

Reconstruction of the refueling history during the early years of operation has been difficult using data readily accessible to BNL Staff (direct access to the Licensee for information was precluded). Table A.11 lists the refueling data used by BNL for the ORIGEN2 calculations, which were carried out in 1985.

Subsequently, additional information has been located that would permit a revision of the data in Table A.11, but repeating the ORIGEN2 calculations did not seem worthwhile since only minor changes in the spent fuel radioactive inventories would have resulted. At the time Table A.11 was constructed, no data on the first refueling in February, 1971 was available.[13] Also, some 84 fuel assemblies from early refuelings could not be accounted for. Later, it was learned that 81 assemblies had been shipped for reprocessing at the West Valley facility. These apparently were returned in 1985 to Ginna for storage in the spent fuel pool.[13]

At the time of the April 1972 refueling, cladding distortions due to fuel densification was discovered and 61 assemblies were replaced (Ref. 12, pg. F-56). Thus, the entry in Table A.11 for the second discharge is incorrect.

The total burnup not accounted for in the ORIGEN2 calculations amounts to 4.2% of the total thermal energy production from 1969 through April 1, 1987. The missing 4.2% burnup is for fuel discharged on or before April 1972.

A.4.3 PWR Fuel Assembly Model Used in ORIGEN2 Calculations

A nominal PWR fuel element has been modeled based on data presented in Ref. 9. This is a 17x17 element assembly (264 fuel elements per assembly) of 3.2% U-235 enrichment containing 461.4 kg of uranium. The cladding is Zircaloy-4. Other alloys present in the fuel assembly hardware include Inconel-718, Nicrobraze 50, SS-302 and SS-304. The alloy contents of the assembly hardware are included with weighting factors to take account of the axial variation of the neutron flux which results in lower neutron flux which results in lower neutron activation at the ends of the assemblies. In addition to the fuel, the cladding and the assembly hardware, an allowance was made for the presence of "crud," composed of Cr, Fe, Co and Ni, on the outer surfaces of the cladding and hardware.

No corrections were made in the ORIGEN2 calculations to account for stainless steel clad fuel that was used in the early history of the plant.

A.4.4 Calculated Radioactive Inventories

The calculated inventories of selected radionuclides* are listed in Table A.12 for the end of operating cycle number 16 projected to be on April 1, 1987. Also listed are the inventories in the spent fuel basin on May 1 and July 1, 1987 and April 1, 1988, assuming that 24 assemblies will be discharged in the April 1987 refueling.

It should be noted that many of the isotopes that are of considerable importance in a core melt accident are those of short half-lives which are no longer present in the spent fuel after a few days of decay, e.g., Rb-91, Rb-93, Sr-93, Sr-95, Y-94, Y-95, Tc-104, I-134, I-135, I-136, Cs-138, Cs-140. On the other hand, the spent fuel inventory contains much larger quantities of several long-lived isotopes than does the equilibrium core. Noteworthy among these are H-3, C-14, Sr-90 (Y-90), I-129, Cs-137, Ba-137m, Eu-154, Pu-239, Pu-240, Pu-241, Am-241, and Cm-244.

*The selection of radionuclides was based on several criteria including potential for biological concern, thermal power and total curies of activity.

Table A.13 gives a comparison of the radionuclide inventories in the last fuel batch to be discharged with the summation of the inventories contained in batches 2-15 discharged between 1976 and 1986.

A.4.5 Decay Heat

Table A.14 summarizes the decay heat production in the various discharged batches. The data shown is for the whole batch, i.e., the specific thermal power (kilowatts per metric tonne) multiplied by the metric tonnes in the batch.

Table A.15 summarizes the fraction of the decay heat contributed by various isotopes. The main contributors change with decay time, e.g., in the oldest fuel (batches 2, 3, etc.) the largest contributors are Y-90 and Ba-137m, whereas the last discharged batch 16 is dominated by Cs-134, Rh-106 and Pr-144. The actinides are relatively small contributors.

A.5 REFERENCES FOR APPENDIX A

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6. A.G. Croff, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials," Nuclear Technology. Vol. 62, pp. 335-352, September 1983.

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9. A.G. Croff, M.A. Bjerke, G.W. Morrison, and L.M. Petrie, Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code, Oak Ridge National Laboratory, ORNL/TM-6051, September 1978.

10. U.S. Nuclear Regulatory Commission, Integrated Plant Safety Assessment Systematic Evaluation Program, Millstone Nuclear Power Station, Unit 1, NUREG-0824, February 1983.

11. U.S. Nuclear Regulatory Commission, Nuclear Power Plant Operating Experience, 1974-1975, NUREG-0227, April 1977, pg. 65.

12. U.S. Nuclear Regulatory Commission, Integrated Plant Safety Assessment Systematic Evaluation Program, R.E. Ginna Nuclear Power Plant, NUREG-0821, December 1982.

13. Rochester Gas and Electric Corporation, "Application for Amendment to Operating License to Amend Appendix A to Increase Spent Fuel Pool Storage Capacity," submitted to NRC April 2, 1984, Docket No.50-244.

Table A.1 Reactor and Fuel Cycle Parameters for Millstone 1 (Sources: Refs. 1-4)

Assemblies in core:	580
Licensed thermal power:	2011 MW_{th} (gross)
Thermal power corresponding to maximum dependable capacity:	2006.5 MW_{th} (gross)
Nominal initial metric tonnes of heavy metal (IMTHM) per assembly:	0.1833 MT
Average refueling cycle interval (since initial commercial operation):	21 to 22 months
Recent refueling cycle interval (since April, 1979):	about 18 months
Average number of assemblies per discharge:	about 173
Average IMTHM per discharge:	about 31.7 MT
Average number of fuel cycles per assembly:	about 3.35
Average period of irradiation (including downtime):	about 72 months
Authorized Storage Pool Capacity (as of 1985):	2184 assemblies

Table A.2 Summary of Operational Milestones for Millstone 1 (Source: Ref. 4)

Date of Initial Criticality: October 26, 1970
Date of First Electricity Generation: November 29, 1970
Date of Commercial Operation: March 1, 1971
Lifetime Cumulative Data: (January 1, 1971 - March 31, 1986)
Hours, Generator on Line: 100,307.9 hours
Gross Thermal Energy: 184.83×10^6 MWh
Capacity Factor (MDC net): 67.4%

Table A.3 Summary of Spent Fuel Batches in Millstone 1 Storage Basin (With Projections to 1987)

Spent Fuel Batch No.	Date of End of Irradiation	Number of Assemblies	Weight H.M. (MT)	Avg. Burnup (MWD/MT)	Decay[a] Days to 5/1/87 (days)	Cumulative Assemblies in Pool	Cumulative[b] Gross Weight of Spent Fuel in Pool (MT)
1	08/31/72	28	5.132	12686	5356	28	8.95
2	08/31/74	204	38.126	19695	4626	236	75.47
3	09/11/75	144	26.395	26581	4250	380	121.52
4	09/30/76	124	22.729	21290	3865	504	161.18
5	03/10/78	124	22.729	24090	3339	628	200.83
6	04/27/79	148	27.128	24354	2926	776	248.16
7	10/03/80	168	30.794	24998	2394	944	301.89
8	09/11/82	192	35.194	23670	1693	1136	363.29
9	04/12/84	172	31.528	26763	1114	1308	418.30
10[c]	10/01/85	178	32.627	28052	577	1486	475.22
11[c]	04/01/87	167	30.611	29963	30	1653[d]	528.63

[a]Decay days from end of irradiation to 5/1/87.
[b]Gross fuel tonnage in pool includes heavy metal plus cladding and hardware but not including fuel racks. Each assembly contains approximately 0.1833 metric tonnes of heavy metal, 0.0246 tonnes of oxygen (in UO_2) and 0.1119 tonnes of hardware, totaling 0.3198 metric tonnes gross.
[c]Projected data.
[d]The present authorized storage capacity is 2184 assemblies. After the 04/01/87 refueling, the accumulated assemblies plus the 580 assemblies in the core would exceed the authorized storage capacity should a full core discharge be required.

Table A.4 Comparison of Cumulative Gross Thermal Energy Production with Calculated Fuel Burnup from Start of Operations in 1970 to April 1, 1987 (Millstone 1)

Total Cumulative Gross Thermal Energy (MWD x 10^{-3})	Spent Fuel Batch No.	Total Burnup in Batch (MWD x 10^{-3})
	1	65.10
	2	750.88
	3	701.61
	4	483.91
	5	547.54
	6	660.68
	7	769.78
	8	833.05
	9	843.78
	10	915.25
	11	917.21
	12*	612.74
	13*	329.55
Total 8440.25		8440.01

*Burnup in fuel remaining in the core.

Table A.5 Comparison of Radioactive Inventories in Reactor Core and Spent Fuel Basin (Millstone 1). The Assumed Refueling Scenario is Described in Section A.3.4

Isotope	Reactor Core	Spent Fuel Storage Basin[a]			
		5/1/87	7/1/87	10/1/87	4/1/88
	(Radioactivity, Curies)				
H 3	4.95E+4	1.38E+5	1.37E+5	1.35E+5	1.31E+5
C 14	1.02E+2	4.12E+2	4.12E+2	4.12E+2	4.12E+2
Co 58	8.81E+4	2.29E+4	1.26E+4	5.12E+3	8.54E+2
Co 60	1.64E+5	3.72E+5	3.15E+5	3.04E+5	2.85E+5
Kr 85	5.35E+5	1.41E+6	1.39E+6	1.37E+6	1.33E+6
Rb 86	6.22E+4	1.01E+4	1.05E+3	3.44E+1	3.84E-2
Sr 89	4.71E+7	8.39E+6	3.63E+6	1.03E+6	8.33E+4
Sr 90	4.25E+6	1.42E+7	1.42E+7	1.41E+7	1.39E+7
Y 90	4.37E+6	1.43E+7	1.42E+7	1.41E+7	1.39E+7
Y 91	6.06E+7	1.18E+7	5.75E+6	1.98E+6	2.21E+5
Zr 95	8.70E+7	1.94E+7	1.00E+7	3.70E+6	5.10E+5
Nb 95	8.91E+7	2.54E+7	1.70E+7	7.35E+6	1.11E+6
Mo 99	8.78E+7	1.49E+4	3.12E-3	neg.[b]	neg.[b]
Tc 99m	7.69E+7	1.43E+4	3.01E-3	neg.[b]	neg.[b]
Ru 103	7.23E+7	1.53E+7	5.21E+6	1.03E+6	4.07E+4
Ru 106	2.48E+7	1.72E+7	1.53E+7	1.29E+7	9.13E+6
Rh 106	2.63E+7	1.72E+7	1.53E+7	1.29E+7	9.13E+6
Sb 125	9.07E+5	1.19E+6	1.14E+6	1.07E+6	9.48E+5
Sb 127	4.97E+6	8.21E+3	1.39E-1	neg.[b]	neg.[b]
Te 125m	1.93E+5	2.84E+5	2.76E+5	2.61E+5	2.31E+5
Te 127	4.92E+6	2.21E+5	1.45E+5	8.06E+4	2.52E+4
Te 127m	6.61E+5	2.18E+5	1.48E+5	8.23E+4	2.57E+4
Te 129	1.49E+7	2.74E+5	7.79E+4	1.17E+4	2.68E+2
Te 129m	2.24E+6	4.21E+5	1.20E+5	1.79E+4	4.12E+2
Te 132	6.72E+7	3.74E+4	8.64E-2	neg.[b]	neg.[b]
I 129	1.75E+0	7.15E+0	7.15E+0	7.15E+0	7.15E+0
I 131	4.74E+7	1.22E+6	6.35E+3	2.28E+0	neg.[b]
I 132	6.83E+7	3.85E+4	8.90E-2	neg.[b]	neg.[b]
Xe 133	9.72E+7	7.29E+5	2.30E+2	1.21E-3	neg.[b]
Cs 134	6.10E+6	7.90E+6	7.47E+6	6.86E+6	5.80E+6
Cs 136	2.10E+6	2.05E+5	8.13E+3	6.26E+1	3.91E-3
Cs 137	5.84E+6	2.02E+7	2.01E+7	2.00E+7	1.97E+7
Ba 137m	5.53E+6	1.91E+7	1.90E+7	1.89E+7	1.87E+7
Ba 140	8.36E+7	5.19E+6	1.90E+5	1.30E+3	6.41E-2
La 140	8.54E+7	5.97E+6	2.19E+5	1.50E+3	7.37E-2
Ce 141	7.94E+7	1.32E+7	3.61E+6	5.07E+5	1.03E+4
Ce 144	6.05E+7	2.64E+7	2.27E+7	1.81E+7	1.16E+7
Pr 143	7.37E+7	5.44E+6	2.41E+5	2.19E+3	1.90E-1
Pr 144	6.08E+7	2.64E+7	2.27E+7	1.81E+7	1.16E+7
Nd 147	3.16E+7	1.54E+6	3.36E+4	1.05E+2	1.10E-3
Sm 151	2.44E+4	8.22E+4	8.21E+4	8.19E+4	8.16E+4
Eu 154	4.61E+5	1.34E+6	1.32E+6	1.29E+6	1.25E+6
Eu 156	5.61E+6	8.26E+5	5.10E+4	7.76E+2	1.80E-1
Np 239	9.98E+8	5.59E+4	2.88E+3	2.88E+3	2.88E+3
Pu 238	9.33E+4	4.51E+5	4.53E+5	4.54E+5	4.54E+5
Pu 239	2.49E+4	8.89E+4	8.89E+4	8.89E+4	8.89E+4
Pu 240	3.14E+4	1.30E+5	1.30E+5	1.30E+5	1.30E+5
Pu 241	7.19E+6	2.29E+7	2.27E+7	2.25E+7	2.19E+7
Am 241	8.86E+3	2.88E+5	2.94E+5	3.03E+5	3.21E+5
Cm 242	2.09E+6	1.45E+6	1.12E+6	7.60E+5	3.50E+5
Cm 244	6.72E+4	2.27E+5	2.25E+5	2.23E+5	2.19E+5

[a]Spent fuel pool inventory includes discharges from 11 refuelings covering the period from August 1972 through the projected refueling of April 1987.

[b]neg. = less than 10^{-3} Curies.

Table A.6 Comparison of Radioactive Inventories of Most Recently Discharged Fuel Batch (Batch 11) with Longer Aged Discharged Batches (Batches 1-10) (Millstone 1)

Isotope	Spent Fuel Batch 11[a]				Spent Fuel Batch 1-10[b]			
	5/1/87	7/1/87	10/1/87	4/1/88	5/1/87	7/1/87	10/1/87	4/1/88
	(Radioactivity, Curies)							
H 3	2.27E+4	2.24E+4	2.21E+4	2.15E+4	1.16E+5	1.15E+5	1.13E+5	1.10E+5
C 14	5.18E+1	5.18E+1	5.18E+1	5.18E+1	3.61E+2	3.61E+2	3.61E+2	3.61E+2
Co 58	2.28E+4	1.26E+4	5.10E+3	8.49E+2	1.18E+2	6.48E+1	2.63E+1	4.39E+0
Co 60	7.64E+4	7.48E+4	7.24E+4	6.77E+4	2.45E+5	2.40E+5	2.32E+5	2.17E+5
Kr 85	2.39E+5	2.36E+5	2.32E+5	2.25E+5	1.17E+6	1.16E+6	1.14E+6	1.10E+6
Rb 86	1.01E+4	1.05E+3	3.44E+1	3.84E-2	neg.[c]	neg.	neg.	neg.
Sr 89	8.39E+6	3.63E+6	1.03E+6	8.33E+4	5.39E+3	2.33E+3	6.60E+2	5.35E+1
Sr 90	1.93E+6	1.92E+6	1.91E+6	1.89E+6	1.23E+7	1.23E+7	1.22E+7	1.21E+7
Y 90	1.93E+6	1.92E+6	1.91E+6	1.89E+6	1.23E+7	1.23E+7	1.22E+7	1.21E+7
Y 91	1.18E+7	5.74E+6	1.93E+6	2.21E+5	2.11E+4	1.02E+4	3.44E+3	3.94E+2
Zr 95	1.94E+7	1.00E+7	3.69E+6	5.09E+5	5.90E+4	3.05E+4	1.12E+4	1.55E+3
Nb 95	2.53E+7	1.69E+7	7.33E+6	1.11E+6	1.31E+5	6.76E+4	2.49E+4	3.44E+3
Mo 99	1.49E+4	3.12E-3	neg.	neg.	neg.	neg.	neg.	neg.
Tc 99m	1.43E+4	3.01E-3	neg.	neg.	neg.	neg.	neg.	neg.
Ru 103	1.53E+7	5.21E+6	1.03E+6	4.07E+4	1.09E+3	3.73E+2	7.35E+1	2.91E+0
Ru 106	1.12E+7	9.98E+6	8.40E+6	5.95E+6	5.98E+6	5.30E+6	4.48E+6	3.18E+6
Rh 106	1.12E+7	9.98E+6	8.40E+6	5.95E+6	5.98E+6	5.30E+6	4.48E+6	3.18E+6
Sb 125	4.17E+5	4.00E+5	3.76E+5	3.31E+5	7.76E+5	7.44E+5	6.99E+5	6.16E+5
Sb 127	8.21E+3	1.39E-1	neg.	neg.	neg.	neg.	neg.	neg.
Te 125m	9.42E+4	9.39E+4	9.04E+4	8.07E+4	1.89E+5	1.82E+5	1.70E+5	1.50E+5
Te 127	2.14E+5	1.40E+5	7.79E+4	2.43E+4	7.15E+3	4.85E+3	2.70E+3	8.44E+2
Te 127m	2.10E+5	1.43E+5	7.95E+4	2.48E+4	7.30E+3	4.95E+3	2.76E+3	8.62E+2
Te 129	2.74E+5	7.79E+4	1.17E+4	2.68E+2	3.85E+0	1.09E+0	1.64E-1	3.76E-3
Te 129m	4.21E+5	1.20E+5	1.79E+4	4.12E+2	5.91E+0	1.68E+0	2.52E-1	5.77E-3
Te 132	3.74E+4	8.64E-2	neg.	neg.	neg.	neg.	neg.	neg.
I 129	8.84E-1	8.86E-1	8.86E-1	8.86E-1	6.26E+0	6.26E+0	6.26E+0	6.26E+0
I 131	1.22E+6	6.35E+3	2.28E+0	neg.	neg.	neg.	neg.	neg.
I 132	3.85E+4	8.90E-2	neg.	neg.	neg.	neg.	neg.	neg.
Xe 133	7.29E+5	2.30E+2	1.21E-3	neg.	neg.	neg.	neg.	neg.
Cs 134	3.53E+6	3.34E+6	3.07E+6	2.59E+6	4.37E+6	4.13E+6	3.80E+6	3.21E+6
Cs 136	2.05E+5	8.13E+3	6.26E+1	3.91E-3	neg.	neg.	neg.	neg.
Cs 137	2.83E+6	2.82E+6	2.80E+6	2.77E+6	1.73E+7	1.73E+7	1.72E+7	1.70E+7
Ba 137m	2.67E+6	2.66E+6	2.65E+6	2.62E+6	1.64E+7	1.63E+7	1.63E+7	1.61E+7
Ba 140	5.19E+6	1.90E+5	1.30E+3	6.41E-2	neg.	neg.	neg.	neg.
La 140	5.97E+6	2.19E+5	1.50E+3	7.37E-2	neg.	neg.	neg.	neg.
Ce 141	1.32E+7	3.61E+6	5.07E+5	1.03E+4	1.31E+2	3.57E+1	5.02E+0	1.01E-1
Ce 144	1.91E+7	1.65E+7	1.32E+7	8.43E+6	7.23E+6	6.23E+6	4.98E+6	3.19E+6
Pr 143	5.44E+6	2.41E+5	2.19E+3	1.90E-1	neg.	neg.	neg.	neg.
Pr 144	1.91E+7	1.65E+7	1.32E+7	8.43E+6	7.23E+6	6.23E+6	4.98E+6	3.19E+6
Nd 147	1.54E+6	3.36E+4	1.05E+2	1.10E-3	neg.	neg.	neg.	neg.
Sm 151	9.31E+3	9.30E+3	9.28E+3	9.25E+3	7.29E+4	7.28E+4	7.26E+4	7.24E+4
Eu 154	2.89E+5	2.85E+5	2.79E+5	2.68E+5	1.05E+6	1.04E+6	1.02E+6	9.75E+5
Eu 156	8.38E+5	5.18E+4	7.76E+2	1.83E-1	neg.	neg.	neg.	neg.
Np 239	5.36E+4	5.26E+2	5.26E+2	5.26E+2	2.35E+3	2.35E+3	2.35E+3	2.35E+3
Pu 238	6.73E+4	6.87E+4	7.02E+4	7.18E+4	3.84E+5	3.84E+5	3.83E+5	3.82E+5
Pu 239	9.28E+3	9.28E+3	9.28E+3	9.28E+3	7.96E+4	7.96E+4	7.96E+4	7.96E+4
Pu 240	1.55E+4	1.55E+4	1.55E+4	1.55E+4	1.15E+5	1.15E+5	1.15E+5	1.15E+5
Pu 241	3.73E+6	3.70E+6	3.65E+6	3.56E+6	1.92E+7	1.91E+7	1.88E+7	1.84E+7
Am 241	6.01E+3	7.00E+3	8.48E+3	1.14E+4	2.82E+5	2.87E+5	2.95E+5	3.09E+5
Cm 242	1.31E+6	1.01E+6	6.86E+5	3.16E+5	1.39E+5	1.08E+5	7.33E+4	3.47E+4
Cm 244	5.88E+4	5.84E+4	5.79E+4	5.68E+4	1.68E+5	1.67E+5	1.65E+5	1.62E+5

[a]Fuel batch 11 is projected discharge during April 1987.
[b]Fuel batches 1-10 were discharged between August 1972 and October 1985.
[c]neg. = less than 10^{-3} Curies.

Table A.7 Decay Heat Released from Spent Fuel Inventory for Various Discharged Fuel Batches (Millstone 1)

	Date, End of Irradiation[a]	Batch Size[a] (Metric Tonnes)	Decay Heat Released by Batch (Kilowatts, Thermal) May 1, 1987	July 1, 1987	April 1, 1988
1	08/31/72	5.13	1.8	1.8	1.8
2	08/31/74	38.13	22.0	21.9	21.5
3	09/11/75	26.40	21.8	21.7	21.2
4	09/30/76	22.73	15.2	15.1	14.8
5	03/10/78	22.73	18.4	18.3	17.7
6	04/27/79	27.13	23.5	23.3	22.4
7	10/03/80	30.79	30.3	29.9	28.2
8	09/11/82	35.19	41.5	40.3	35.9
9	04/12/84	31.53	67.4	63.6	50.9
10	10/01/85[b]	32.63	146.0	132.7	91.8
11	04/01/87[b]	30.61	909.0	537.7	210.5
Total[c] 1-10		272.38	387.9	368.5	306.3
Total[c] 1-11		302.99	1297.0	906.3	516.8

[a]See Table A.3.
[b]Projected dates.
[c]Totals may not equal sum of the entries due to rounding of decimals.

Table A.8 Radionuclide Contributions to Decay Heat for Various Spent Fuel Batches. The Percentage Contributions Depend on the Total Burnup of Each Batch, as well as Decay Time After End of Irradiation (Millstone 1)

Isotope	Spent Fuel Batch Number										
	1	2	3	4	5	6	7	8	9	10	11
	(PERCENT OF TOTAL DECAY HEAT)										
Sr 90	7.48	6.79	6.14	6.61	6.32	6.18	5.85	5.23	3.78	2.34	1.39
Y 90	35.73	32.44	29.33	31.82	30.21	29.52	27.92	24.97	18.06	11.17	4.96
Zr 95	---[a]	---	---	---	---	---	---	---	---	0.01	1.13
Nb 95	---	---	---	---	---	---	---	---	---	0.02	2.33
Rh 106	---	---	---	---	---	0.81	1.89	5.75	13.49	22.53	27.10
Cs 134	0.43	0.98	1.76	2.24	3.74	5.15	7.65	11.63	16.26	16.66	12.53
Cs 137	9.02	8.77	8.43	8.70	8.45	8.28	7.87	6.95	5.16	3.22	1.45
Ba 137m	30.29	29.44	28.30	29.22	28.29	27.80	26.43	23.34	17.34	10.82	4.88
Ce 144	---	---	---	---	---	---	0.06	0.26	0.79	1.73	2.66
Pr 144	---	---	---	---	---	---	0.64	2.93	8.80	19.20	29.42
Eu 154	1.22	2.15	3.03	2.63	3.15	3.32	3.52	3.30	3.03	2.15	1.12
Pu 238	2.14	4.85	7.33	4.66	5.38	5.31	5.33	4.49	3.72	2.37	1.13
Pu 239	2.16	1.54	1.14	1.36	1.16	1.10	0.99	0.88	0.57	0.33	0.14
Pu 240	1.84	1.90	1.79	1.78	1.68	1.61	1.49	1.27	0.90	0.53	0.23
Pu 241	0.19	0.22	0.23	0.24	0.24	0.24	0.24	0.22	0.17	0.11	0.05
Am 241	7.57	7.96	7.34	6.70	5.84	5.12	4.22	2.92	1.61	0.70	0.18
Cm 242	0.01	0.04	0.05	0.02	0.03	0.03	0.03	0.05	0.21	1.21	5.52
Totals[b]	98.08	97.08	94.87	95.98	94.59	94.47	94.13	94.19	93.89	95.10	96.22

[a]Dashes indicate less than 0.01%.

[b]Total percentage of isotopes listed. The balance of the decay heat is distributed among many other less important contributors.

Table A.9 Reactor and Fuel Cycle Parameters for Ginna
(Sources: Refs. 1-4)

Assemblies in core: 121			
Licensed thermal power: 1520 MW_{th} (gross)[a]			
Thermal power corresponding to maximum dependable capacity: 1499 MW_{th} (gross)			
Nominal initial metric tonnes of heave metal (IMTHM) per assembly: 0.375 MT			
Average refueling cycle interval (since initial commercial operation): 12.6 months			
Average number of assemblies per discharge:	1975-1980:	37	
	1981-1987:	24	
Average IMTHM per discharge:	1975-1908:	15.3 MT	
	1981-1987:	9.0	
Average number of fuel cycles per assembly:	1975-1980:	3.27	
	1981-1987:	5.04	
Average period of irradiation (including down time):	1976-1980:	3.3 years	
	1981-1987:	5.0 years.	
Authorized storage pool capacity: 1016			

[a]On March 1, 1972 the Atomic Energy Commission authorized an increase in gross thermal power from 1300 to 1520 MW.

Table A.10 Summary of Operational Milestone for Ginna (Source: Ref. 4)

Date of Initial Criticality: November 8, 1969
Date of First Electricity Generation: December 2, 1969
Date of Commercial Operation: July 1, 1970
Lifetime Cumulative Data: (January 1, 1968-March 31, 1986)
Hours, Generator on Line: 107,134.3 hours
Gross Thermal Energy: 149.26 x 10^6 MWh
Capacity Factor (MDC net): 70.3%

Table A.11 Summary of Spent Fuel Batches in Ginna Storage Basin (With Projections to 1987)

Spent Fuel Batch No.	Date of End of Irradiation	Number of Assemblies	Weight H.M. (MT)	Avg. Burnup (MWD/MT)	Decay[a] Days to 5/1/87 (days)	Cumulative Assemblies in Pool	Cumulative[b] Gross Weight of Spent Fuel in Pool (MT)
1	02/27/71	(37)[c]	14.778	6933	--	0	0
2	04/13/72	(47)[c]	18.772	16695	5832	28	18.4
3	12/31/73	8	3.195	30039	4869	36	23.7
4	03/08/75	29	11.583	38043	4437	65	42.8
5	01/28/76	37	14.778	36958	4111	102	67.1
6	04/14/77	41	16.375	36022	3669	143	94.1
7	03/23/78	41	16.375	27921	3326	184	121.1
8	02/09/79	40	15.976	25451	3003	224	147.4
9	03/28/80	36	14.378	26088	2590	260	171.1
10	04/17/81	28	11.183	27884	2205	288	189.5
11	01/25/82	24	9.586	31054	1891	312	205.3
12	03/25/83	20	7.988	33772	1467	332	218.4
13	03/01/84	23	9.186	37532	1156	355	223.6
14	02/28/85	25	9.985	40533	792	380	250.0
15	03/30/86	24	9.586	42360	397	404	265.8
16[d]	04/01/87	24	9.586	45673	30	428[e]	281.6

[a]Decay days from end of irradiation to 5/1/87.
[b]Gross weight of fuel stored in pool includes heavy metal plus cladding and hardware but not the fuel racks. Each assembly contains approximately 0.4614 tonnes of heavy metal, 0.0620 tonnes of oxygen, 0.1345 tonnes of hardware, totaling 0.6579 tonnes gross.
[c]At the time of the ORIGEN2 calculations some 56 assemblies could not be accounted for using available data.
[d]Projected data.
[e]Authorized capacity is 1016 assemblies.

Table A.12 Comparison of Radioactive Inventories in Reactor Core and Spent Fuel Basin (Ginna)

Isotope	Reactor Core	Spent Fuel Storage Basin[a]		
		5/1/87	7/1/87	4/1/88
	(Radioactivity, Curies)			
H 3	3.32E+4	9.29E+4	9.20E+4	8.82E+4
C 14	6.42E+1	2.64E+2	2.64E+2	2.64E+2
Co 58	3.57E+5	5.93E+4	3.26E+4	2.21E+3
Co 60	3.20E+5	5.97E+5	5.84E+5	5.29E+5
Kr 85	3.73E+5	9.84E+5	9.74E+5	9.27E+5
Rb 86	6.53E+5	7.22E+3	7.48E+2	2.74E-2
Sr 89	3.55E+7	3.53E+6	1.33E+6	3.50E+4
Sr 90	2.95E+6	1.02E+7	1.01E+7	9.95E+6
Y 90	3.15E+6	1.02E+7	1.01E+7	9.95E+6
Y 91	4.57E+7	5.11E+6	2.48E+6	9.54E+4
Zr 95	6.41E+7	8.64E+6	4.46E+6	2.27E+5
Nb 95	6.34E+7	1.12E+7	7.51E+6	4.93E+5
Mo 99	6.83E+7	7.03E+3	1.48E-3	neg.[b]
Tc 99m	5.89E+7	6.77E+3	1.42E-3	neg.[b]
Ru 103	5.85E+7	7.86E+6	2.88E+6	2.09E+4
Ru 106	1.95E+7	1.09E+7	9.71E+6	5.78E+6
Rh 106	2.15E+7	1.09E+7	9.71E+6	5.78E+6
Sb 125	6.04E+5	7.11E+5	6.82E+5	5.65E+5
Sb 127	4.12E+6	4.33E+3	7.35E-2	neg.[b]
Te 125m	1.27E+5	1.70E+5	1.65E+5	1.37E+5
Te 127	4.05E+6	1.19E+5	7.79E+4	1.36E+4
Te 127m	5.19E+5	1.17E+5	7.95E+4	1.38E+4
Te 129	1.21E+7	1.38E+5	3.93E+4	1.35E+2
Te 129m	1.80E+6	2.12E+5	6.03E+4	2.07E+2
Te 132	5.33E+7	1.83E+4	4.23E-2	neg.[b]
I 129	1.27E+0	5.32E+0	5.32E+0	5.32E+0
I 131	3.76E+7	6.00E+5	3.12E+3	neg.[b]
I 132	5.42E+7	1.89E+4	4.36E-2	neg.[b]
Xe 133	7.64E+7	3.52E+5	1.11E+2	neg.[b]
Cs 134	5.82E+6	6.35E+6	6.00E+6	4.66E+6
Cs 136	1.87E+6	1.26E+5	4.99E+3	2.40E-3
Cs 137	4.21E+6	1.48E+7	1.47E+7	1.44E+7
Ba 137m	4.00E+6	1.40E+7	1.39E+7	1.37E+7
Ba 140	6.55E+7	2.47E+6	9.07E+4	3.05E-2
La 140	6.74E+7	2.85E+6	1.04E+5	3.51E-2
Ce 141	6.28E+7	6.34E+6	1.72E+6	4.91E+3
Ce 144	4.24E+7	1.38E+7	1.19E+7	6.09E+6
Pr 143	5.71E+7	2.54E+6	1.12E+5	8.86E-2
Pr 144	4.27E+7	1.38E+7	1.19E+7	6.09E+6
Nd 147	2.48E+7	7.42E+5	1.62E+4	neg.[b]
Sm 151	1.42E+4	5.14E+4	5.13E+4	5.10E+4
Eu 154	4.09E+5	1.09E+6	1.07E+6	1.01E+6
Eu 156	7.22E+6	7.58E+5	4.68E+4	1.66E-1
Np 239	7.81E+8	3.02E+4	3.26E+3	3.26E+3
Pu 238	1.01E+5	4.46E+5	4.46E+5	4.46E+5
Pu 239	1.35E+4	5.25E+4	5.25E+4	5.25E+4
Pu 240	2.02E+4	8.60E+4	8.60E+4	8.61E+4
Pu 241	4.85E+6	1.52E+7	1.51E+7	1.46E+7
Am 241	4.99E+3	2.10E+5	2.14E+5	2.32E+5
Cm 242	1.91E+6	9.33E+5	7.20E+5	2.25E+5
Cm 244	1.25E+5	3.59E+5	3.56E+5	3.46E+5

[a]Spent fuel pool inventory includes discharges from 15 refuelings covering the period from April 1983 through the projected refueling of April 1987.

[b]neg. = less than 10^{-3} Curies.

Table A.13 Comparison of Radioactive Inventories of Most Recently Discharged Fuel Batch (Batch 16) with Longer Aged Discharged Batches (Batches 2-15) (Ginna)

Isotope	Spent Fuel Batch 16[a]				Spent Fuel Batch 2-15[b]			
	5/1/87	7/1/87	10/1/87	4/1/88	5/1/87	7/1/87	10/1/87	4/1/88
	(Radioactivity, Curies)							
H 3	9.89E+3	9.80E+3	9.66E+3	9.39E+3	8.29E+4	8.22E+4	8.10E+4	7.88E+4
C 14	2.20E+1	2.20E+1	2.20E+1	2.20E+1	2.42E+2	2.42E+2	2.42E+2	2.42E+2
Co 58	5.77E+4	3.18E+4	1.29E+4	2.15E+3	1.60E+3	8.78E+2	3.57E+2	5.94E+1
Co 60	9.92E+4	9.70E+4	9.39E+4	8.79E+4	4.98E+5	4.87E+5	4.71E+5	4.41E+5
Kr 85	1.07E+5	1.05E+5	1.04E+5	1.00E+5	8.78E+5	8.68E+5	8.54E+5	8.27E+5
Rb 86	7.22E+3	7.48E+2	2.45E+1	2.73E-2	neg.	neg.	neg.	neg.
Sr 89	3.50E+6	1.52E+6	4.29E+5	3.48E+6	2.39E+4	1.04E+4	2.93E+3	2.38E+2
Sr 90	8.56E+5	8.53E+5	8.48E+5	8.38E+5	9.32E+6	9.28E+6	9.23E+6	9.12E+6
Y 90	8.57E+5	8.53E+5	8.48E+5	8.38E+5	9.32E+6	9.28E+6	9.23E+6	9.12E+6
Y 91	5.04E+6	2.45E+6	8.23E+5	9.41E+4	6.86E+4	3.33E+4	1.12E+4	1.28E+3
Zr 95	8.47E+6	4.37E+6	1.62E+6	2.23E+5	1.64E+5	8.48E+4	3.13E+4	5.60E+4
Nb 95	1.09E+7	7.33E+6	3.19E+6	4.83E+5	3.68E+5	1.89E+5	6.96E+4	9.57E+3
Mo 99	7.03E+3	1.48E-3	neg.[c]	neg.	neg.	neg.	neg.	neg.
Tc 99m	6.77E+3	1.42E-3	neg.	neg.	neg.	neg.	neg.	neg.
Ru 103	7.86E+6	2.68E+6	5.28E+5	2.09E+4	1.18E+4	4.02E+3	7.93E+2	3.14E+1
Ru 106	5.82E+6	5.19E+6	4.37E+6	3.09E+6	5.06E+6	4.51E+6	3.80E+6	2.69E+6
Rh 106	5.82E+6	5.19E+6	4.37E+5	3.09E+6	5.06E+6	4.51E+6	3.80E+6	2.69E+6
Sb 125	1.84E+5	1.76E+5	1.65E+5	1.46E+5	5.28E+5	5.06E+5	4.75E+5	4.19E+5
Sb 127	4.33E+3	7.35E-2	neg.	neg.	neg.	neg.	neg.	neg.
Te 125m	4.13E+4	4.12E+4	3.97E+4	3.55E+4	1.78E+5	1.24E+5	1.16E+5	1.02E+4
Te 127	1.08E+5	7.05E+4	3.93E+4	1.23E+4	1.09E+4	7.42E+3	4.14E+3	1.29E+4
Te 127m	1.06E+5	7.19E+4	4.01E+4	1.25E+4	1.12E+4	7.58E+3	4.22E+3	1.32E+3
Te 129	1.38E+5	3.93E+4	5.88E+3	1.35E+2	6.98E+1	1.98E+1	2.97E+0	6.82E-2
Te 129m	2.12E+5	6.03E+4	9.04E+3	2.07E+2	1.07E+2	3.05E+1	4.57E+0	1.05E-1
Te 132	1.83E+4	4.23E-2	neg.	neg.	neg.	neg.	neg.	neg.
I 129	4.22E-1	4.23E-1	4.23E-1	4.23E-1	4.89E+0	4.89E+0	4.89E+0	4.89E+0
I 131	6.00E+5	3.12E+3	1.12E+0	neg.	neg.	neg.	neg.	neg.
I 132	1.89E+4	4.36E-2	neg.	neg.	neg.	neg.	neg.	neg.
Xe 133	3.52E+5	1.11E+2	neg.	neg.	neg.	neg.	neg.	neg.
Cs 134	2.26E+6	2.13E+6	1.96E+6	1.66E+6	4.09E+6	3.87E+6	3.55E+6	3.01E+6
Cs 136	1.26E+5	4.99E+3	3.84E+1	2.40E-3	neg.	neg.	neg.	neg.
Cs 137	1.34E+6	1.34E+6	1.33E+6	1.31E+6	1.34E+7	1.34E+7	1.33E+7	1.31E+7
Ba 137m	1.27E+6	1.26E+6	1.26E+6	1.24E+6	1.27E+7	1.26E+7	1.26E+7	1.24E+7
Ba 140	2.47E+6	9.07E+4	6.19E+2	3.05E-2	neg.	neg.	neg.	neg.
La 140	2.85E+6	1.04E+5	7.13E+2	3.51E-2	neg.	neg.	neg.	neg.
Ce 141	6.34E+6	1.73E+6	2.43E+5	4.91E+3	2.53E+3	6.89E+2	9.69E+1	1.96E+0
Ce 144	8.25E+6	7.11E+6	5.68E+6	3.64E+6	5.58E+6	4.81E+6	3.84E+6	2.46E+6
Pr 143	2.54E+6	1.12E+5	1.02E+3	8.86E-2	neg.	neg.	neg.	neg.
Pr 144	8.25E+6	7.11E+6	5.68E+6	3.64E+6	5.58E+6	4.81E+6	3.84E+6	2.46E+6
Nd 147	7.42E+5	1.62E+4	5.08E+1	neg.	neg.	neg.	neg.	neg.
Sm 151	3.47E+3	3.47E+3	3.46E+3	3.45E+3	4.79E+4	4.79E+4	4.78E+4	4.76E+4
Eu 154	1.67E+5	1.65E+5	1.61E+5	1.55E+5	9.19E+5	9.06E+5	8.88E+5	8.53E+5
Eu 156	7.58E+5	4.68E+4	7.02E+2	1.66E-1	neg.	neg.	neg.	neg.
Np 239	2.74E+4	4.59E+2	4.59E+2	4.59E+2	2.80E+3	2.80E+3	2.80E+3	2.80E+3
Pu 238	4.87E+4	4.95E+4	5.04E+4	5.13E+4	3.97E+5	3.97E+5	3.96E+5	3.95E+5
Pu 239	3.05E+3	3.05E+3	3.05E+3	3.05E+3	4.95E+4	4.95E+4	4.95E+4	4.95E+4
Pu 240	6.01E+3	6.01E+3	6.01E+3	6.02E+3	8.00E+4	8.00E+4	8.00E+4	8.00E+4
Pu 241	1.58E+6	1.57E+6	1.55E+6	1.51E+6	1.37E+7	1.35E+7	1.34E+7	1.31E+7
Am 241	2.05E+3	2.47E+3	3.10E+3	4.33E+3	2.08E+5	2.12E+5	2.17E+5	2.28E+5
Cm 242	7.57E+5	5.85E+5	3.96E+5	1.82E+5	1.75E+5	1.36E+5	9.21E+4	4.30E+4
Cm 244	8.06E+4	8.00E+4	7.93E+4	7.78E+4	2.78E+5	2.76E+5	2.74E+5	2.68E+5

[a] Fuel batch 16 is projected discharge during April 1987.
[b] Fuel batches 2-15 were discharged between April 1972 and April 1986.
[c] neg. = less than 10^{-3} Curies.

Table A.14 Decay Heat Released from Spent Fuel Inventory for Various Discharged Fuel Batches (Ginna)

	Date, End of Irradiation[a]	Batch Size[a] (Metric Tonnes)	Decay Heat Released by Batch (Kilowatts, Thermal)		
			May 1, 1987	July 1, 1987	April 1, 1988
2	04/13/72	18.772	8.5	8.5	8.4
3	12/31/73	3.195	2.9	2.9	2.8
4	03/08/75	11.583	14.3	14.2	13.9
5	01/28/76	14.778	18.1	18.0	17.6
6	04/14/77	16.375	20.5	20.4	19.8
7	03/23/78	16.375	15.8	15.7	15.1
8	02/09/79	15.976	14.7	14.5	14.0
9	03/28/80	14.378	14.7	14.5	13.7
10	04/17/81	11.183	13.7	13.4	12.4
11	01/25/82	9.586	15.0	14.6	13.2
12	03/25/83	7.988	17.2	16.5	14.2
13	03/01/84	9.186	28.6	27.1	22.0
14	02/28/85	9.985	50.9	47.2	35.3
15	03/30/86	9.586	96.1	85.8	56.5
16	04/01/87[b]	9.586	437.2	260.4	107.7
Total[c] 2-15			331.0	313.3	259.0
Total[c] 2-16			768.3	573.7	366.8

[a]See Table A.11.
[b]Projected dates.
[c]Totals may not equal sum of entries due to rounding of decimals.

Table A.15 Radionuclide Contributions to Decay Heat for Various Spent Fuel Batches. The Percentage Contributions Depend on the Total Burnup of Each Batch, as well as Decay Time After End of Irradiation (Ginna)

Isotope	Spent Fuel Batch Number														
	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
	(PERCENT OF TOTAL DECAY HEAT)														
Sr 90	7.32	6.26	5.56	5.61	5.61	6.23	6.32	6.07	5.61	5.01	4.23	3.43	2.51	1.60	0.90
Y 90	34.95	29.89	26.57	26.82	26.79	29.77	30.20	29.01	26.79	23.95	20.19	16.37	12.01	7.64	4.31
Zr 95	---[a]	---	---	---	---	---	---	---	---	---	---	---	---	0.04	1.01
Nb 95	---	---	---	---	---	---	---	---	---	---	---	---	---	0.08	2.08
Rh 106	---	0.02	0.05	0.08	0.19	0.41	0.76	1.46	2.86	4.65	8.19	12.24	18.16	24.96	27.54
Cs 134	0.33	1.12	1.77	2.26	3.29	4.01	4.95	6.72	9.38	12.21	15.67	18.46	20.18	19.07	15.65
Cs 137	8.89	8.42	7.93	7.94	7.89	8.23	8.19	7.89	7.38	6.75	5.81	4.83	3.63	2.33	1.35
Ba 137m	29.85	28.26	26.64	26.67	26.48	27.65	27.50	26.51	24.79	22.68	19.51	16.22	12.18	7.84	4.52
Ce 144	---	---	---	---	---	0.01	0.02	0.05	0.11	0.19	0.39	0.63	1.07	1.78	2.24
Pr 144	---	---	---	0.01	0.03	0.09	0.23	0.52	1.21	2.13	4.31	6.99	11.84	19.75	24.80
Eu 154	1.33	2.73	3.37	3.48	3.66	3.21	3.05	3.20	3.41	3.64	3.60	3.41	2.88	2.02	1.29
Pu 238	2.90	6.98	9.46	9.12	8.67	5.77	4.76	4.71	4.83	5.18	5.06	4.82	3.95	2.58	1.58
Pu 239	1.84	1.07	0.81	0.82	0.80	1.01	1.05	0.97	0.84	0.69	0.54	0.41	0.28	0.17	0.09
Pu 240	1.80	1.69	1.49	1.49	1.45	1.53	1.47	1.38	1.27	1.13	0.94	0.74	0.53	0.32	0.17
Pu 241	0.17	0.21	0.20	0.20	0.21	0.22	0.22	0.22	0.21	0.20	0.18	0.15	0.11	0.08	0.04
Am 241	7.88	7.68	6.60	6.20	5.56	5.19	4.67	4.07	3.31	2.69	1.97	1.39	0.82	0.37	0.13
Cm 242	0.01	0.03	0.03	0.03	0.03	0.02	0.02	0.02	0.02	0.03	0.07	0.21	0.70	2.20	6.22
Cm244	0.25	2.74	6.45	5.94	5.63	2.18	1.52	1.64	2.03	2.87	3.47	4.25	4.35	3.32	2.52
Totals[b]	97.52	97.10	96.93	96.67	96.29	95.53	94.93	94.44	94.05	94.00	94.13	94.55	95.20	96.15	96.44

[a]Dashes indicate less than 0.01%.
[b]Total percentage of isotopes listed. The balance of the decay heat is distributed among many other less important contributors.

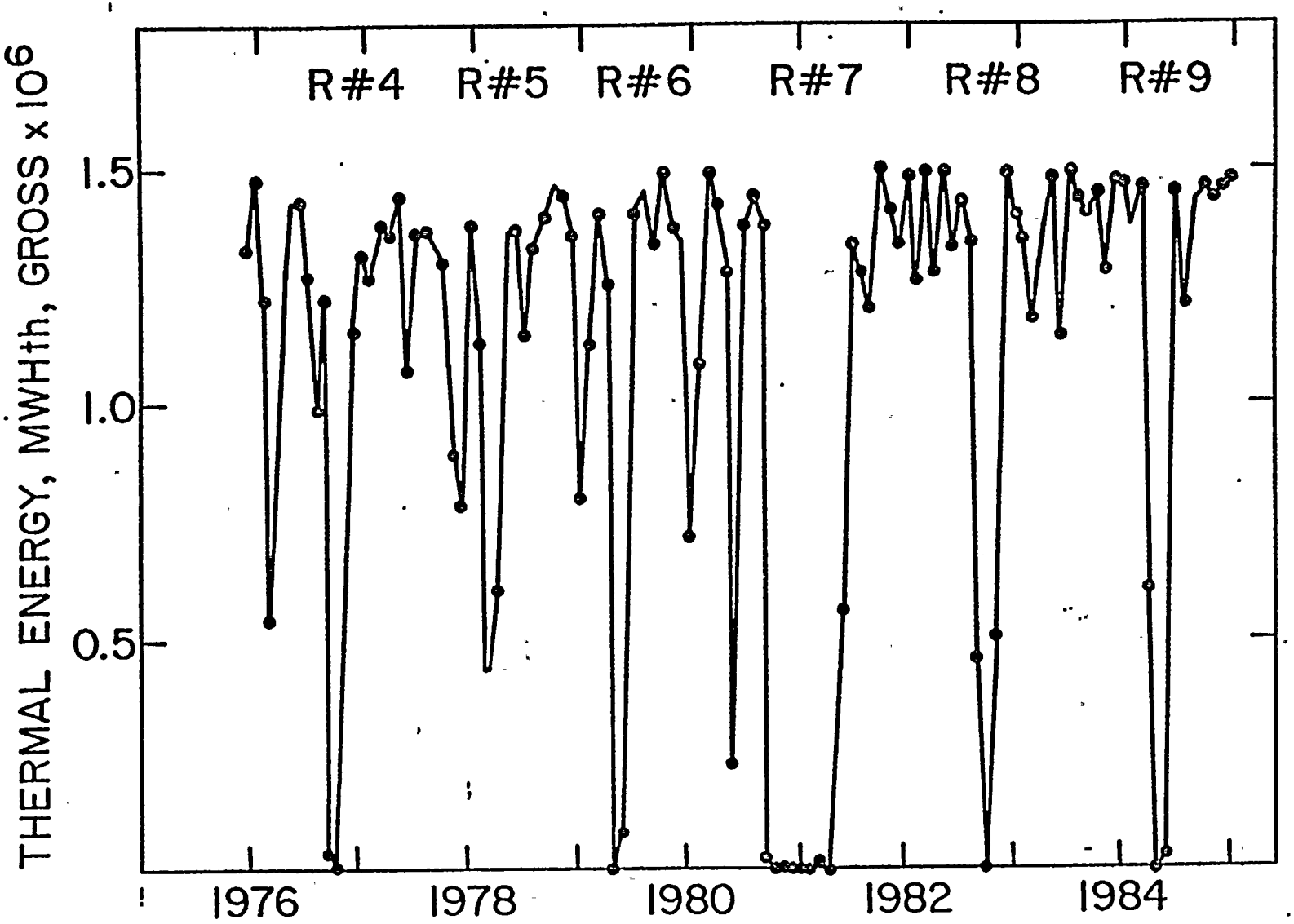

Figure A.1 Millstone 1: Operating history 1976-1984 (Source: Ref. 4).

APPENDIX B

BROOKHAVEN NATIONAL LABORATORY

MEMORANDUM

DATE: August 27, 1986

TO: W.T. Pratt

FROM: K.R. Perkins and H. Connell

SUBJECT: Impact of Revised Reaction Rate Equation on the Likelihood of Zirconium Fires in a Drained Spent Fuel Pool (Task 5)

The SNL investigation[1] of the potential for cladding oxidation during loss of fuel pool inventory accidents has been controversial due to many unique features of the postulated "beyond design basis accident." The purpose of the BNL investigation (FIN A-3786) has been two-fold:

1. Provide an independent assessment of several important areas of the phenomenological treatment of the SFUEL code.[1]

2. Provide an estimate of the likelihood and consequences of the postulated accidents so that the risk can be compared to the risk of severe reactor accidents evaluated in typical PRAs.

The purpose of Task 5 of FIN A-3786 was to re-evaluate the oxidation rate equation used in the SFUEL code and to perform a sensitivity study to demonstrate the influence of the reaction rate on the results of the SFUEL analysis.

The oxidation rate equation is also a key factor which affects the possible propagation of Zircaloy fires to low power (i.e., older) spent fuel bundles. The uncertainty in propagation calculations with SFUEL is addressed in Task 3. A letter report summarizing the results of Task 3 is in preparation and will be submitted to the NRC Project Manager by September 10, 1986.

Discussion

After an extensive review of the zirconium/Zircaloy reaction rate data (Attachment 1) and a second review of some new German data (Attachment 2), we have concluded that the reaction rate used by Benjamin et al.[1] is representative of the existing data. For the purposes of the sensitivity study, we have adopted the two parameter oxidation curve suggested by Weeks. Weeks' two parameter curve is given by:

$$w^2/t = 3.09 \times 10^8 \exp(-56600/RT) \qquad (1)$$

where: w is the oxygen consumption (mg/cm^2)
t is time (sec)
T is the clad temperature (K)
R is the gas constant (1.987 cal/K)

Weeks' equation is equivalent to that suggested by Benjamin[1] except that it provides a smooth transition to the self-sustaining oxidation regime (above 800°C) and does not put undue emphasis on the threshold effect of a shift in oxidation rate due to metallic phase change.

We have varied the reaction rate by a factor of four based on the data scatter in the temperature range of 800 to 900°C (where self-sustaining oxidation is initiated). Only a slight change (±50°C) in the initiation temperature occurs for this broad range of uncertainty in the oxidation rate. This translates into an uncertainty of ±25% in the critical decay power. We believe that this insensitivity to the oxidation rate equation basically confirms the SNL analysis[1] for zirconium fire initiation in a dry spent fuel pool.

As Benjamin et al.[1] pointed out, the most sensitive parameters for clad fire initiation are the decay heat level and the fuel rack geometry (related to natural circulation flow resistance). Thus, for BWRs with low power density and relatively open fuel storage racks, the critical cooling time (to ensure that air cooling will keep the fuel rods below 800°C) is about 1 to 5 months. Whereas PWRs with higher power density and tighter storage racks require 2 months to 2 years (the longer time is required for the new high density storage racks).

Note that even temperatures as low as 650°C can be expected to cause clad failure and release of some fission products if the temperatures are sustained over a long period (several hours). However, below 800°C the energy from oxidation is insufficient to significantly increase the fuel rod temperature.

Conclusions

We conclude that the SNL code (SFUEL) and the clad oxidation rate equation used therein accurately represents the potential for self-sustaining oxidation in a drained fuel pool. The largest uncertainty appears to be due to uncertainties in natural convection flows in the transition flow regime. Changes in the storage rack configuration result in large changes in the calculated flow rate and correspondingly large changes in the "critical power level" (above which self-sustaining oxidation is predicted to occur).

Based on our review of the cladding oxidation rate model and the sensitivity study, we conclude that the conditional probability of self-sustaining clad oxidation and resultant fission product release, given a loss of pool integrity event, is about 10% to 40% for BWRs and 16% to 100% for PWRs, depending on the storage rack configuration.

In terms of power level, our sensitivity studies indicate that the critical power level (above which self-sustaining oxidation will occur) varies from about 50 kW/MTU (for cylindrical racks with large openings) to 6 kW/MTU for the new high density PWR fuel storage racks.

Recommendations

We recommend that spent fuel not be stored in high density racks until it has been stored for 2 or more years in the old style cylindrical racks with adequate coolant openings (3 or more inch diameter holes).

We also recommend that a test program be initiated to confirm the capability of natural air convection cooling capability for high density storage racks. Such tests could be performed with old low power spent fuel (2 to 4 kw/MTU) and minimal instrumentation (such as thermocouples placed near the top of the fuel bundle).

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KRP/HC/csc
Attachments

Attachment 1

BROOKHAVEN NATIONAL LABORATORY

MEMORANDUM

DATE: March 27, 1985

TO: K. Perkins

FROM: L. J. Teutonico

SUBJECT: FIN A-3786 - Study of Beyond Design Basis Accidents in Spent Fuel Pools

Two Sandia reports[1,2] deal with the question of rapid zirconium oxidation in a spent fuel pool following loss of water. Both the computer modeling and the experimental simulation, as described in these reports, suggested that in certain fuel racking configurations (a) a self-sustaining zirconium-air oxidation reaction can be initiated, and (b) this self-sustaining reaction can propagate from one region of a pool to another. There are large uncertainties associated with the phenomenology of zircaloy oxidation and its propagation in spent fuel assemblies. This preliminary report on Tasks 3, 4, 5 of the subject FIN (Uncertainities in Oxidation Propagation, SFUELIW Computer Code Validation, Impact of Revised Reaction Rate Equation, respectively) addresses some of these uncertainities and their effects on the initiation and propagation of a self-sustaining zircaloy-air oxidation reaction.

1) The propagation rates of rapid zircaloy clad oxidation in air from the hottest section of the pool (after a loss of water incident) to adjacent sections were estimated (in Ref. 2) under the conditions that the spent fuel in the hottest section of the pool was generating 30 kw/MTU in a room maintained at constant temperature. As pointed out by Han[3], this estimate should be re-calculated under inadequate room ventilation conditions, to simulate properly the conditions at many licensed facilities. Similarly, additional calculations should be performed in which the hot spent fuel decay power is varied from 20 to 90 kw/MTU for both the adequate and inadequate room ventilation conditions. These studies would determine how sensitive the oxidation propagation is to the decay power of the spent fuel stored adjacent to hot fuel, assuming the input oxidation rate data are known with sufficient accuracy.

2) The above assumes the zircaloy-air reaction rate equation used in the Sandia work is sufficiently accurate. There are a number of uncertainties associated with this equation. We discuss each of these uncertainties in turn.

A. Experimental Data: A literature search [4-147] has revealed that there is a great deal of data for zirconium oxidation; most of it, however, is concerned with oxidation in steam or oxygen. The data for zirconium (zircaloy)-air oxidation presented in Refs. 1 and 2 appear to be the best available. These are shown in Figure 1. The authors (of the SNL reports) fit the data with three separate Arrhenius plots over the temperature range 500-1500°C; one break occurs at the α-β transformation temperature for zirconium, the other at the temperature at which the oxide undergoes a monoclinic-tetragonal transformation. (N.B. two of the sets of data are for zirconium, the other for zircaloy-4). These assumptions are reasonable. It should be noted, however, that there is no a priori reason to expect that the data would be fit by an Arrhenius expression, particularly above the α-β transformation temperature where a number of different processes are occurring simultaneously (discussed further below); therefore the use of the Arrhenius expression should be viewed in this case only as a computational tool. It is difficult to assess the validity of the data employed. What are really required are new experiments to determine the oxidation rate of zircaloy in air over the temperature range of interest, for both isothermal and non-isothermal conditions.

B. Kinetics: The question was raised[2] as to whether the assumption of parabolic kinetics was valid. Data were presented (from Refs. 86 and 126) which show examples of linear as well as cubic kinetics. However, they all apply at temperatures below the α-β transformation temperature. Since almost all rapid oxidation occurs above the α-β transformation temperature, where the oxidation rate is controlled by one or more diffusion processes, the assumption of parabolic kinetics appears to be reasonable.

C. Zirconium vs. Zircaloy: It is assumed in the Sandia work that the oxidation rates of zirconium and zircaloy are essentially the same. Recent work by Pawel and Campbell[136] has shown that this is not the case. Oxidation in steam of both pure zirconium and zircaloy-4 was studied in the temperature range of rapid oxidation (1000°C-1500°C). It was found that at all temperatures the oxidation rate of zircaloy-4 was higher than that of zirconium; the ratio of the two rates is approximately 3 at 1000°C and decreases with increasing temperature to a value of approximately 1.5 at 1500°C (cf. Figure 2). The higher oxidation rate of Zircaloy-4 is attributed to increased oxygen diffusivity in the oxide phase; a lower activation energy was observed, implying that some mechanistic differences exist. Analogous results are expected to apply for oxidation in air.

D. Oxidation Model: The oxidation in steam of both zirconium and zircaloy-4 (in the temperature range 1000-1500°C) is a multi-phase layer process.[136] Not only is an oxide layer formed, but also (beneath it) a layer of oxygen-stabilized α-phase (zirconium or zircaloy). The multi-phase model is only significant above the α-β transformation temperature (approximately 900°C), but this is exactly where rapid oxidation occurs. The parabolic rate constants for oxide layer growth, α-layer growth, and oxygen consumption were determined in Ref. 136 from experimental data and computer modeling. The rate of oxygen consumption is significantly higher at all temperatures than the rate of oxide formation for both zirconium and zircaloy-4. For zirconium the ratio of oxygen consumption rate to oxidation rate is approximately 4 at 1000°C and increases with increasing temperature to a value of approximately 5.4 at 1500°C; for zircaloy-4 the corresponding values are approximately 3.0 and 4.5 at 1000°C and 1500°C, respectively (cf. Figure 2). Although these results were obtained for oxidation in steam, analogous results are again expected for oxidation in air.

E. Effect of Nitrogen: Before discussing the reaction of zirconium with air, let us consider the reaction with nitrogen alone.[148-151] The rate of

reaction of nitrogen with zirconium is much less than the corresponding reaction rate with oxygen; weight gain data[151] after one hour (800°C<T<1200°C) indicate that zirconium reacts with nitrogen about 20 times slower than with oxygen. The overall process is very similar to oxidation in view of the high solubility of nitrogen in zirconium, and involves a large amount of dissolution along with film formation. In the case of nitriding in the α-region, a two phase diffusion process describes the behavior whereas β-phase nitriding involves three phases (nitrogen, like oxygen, stabilizes the α-phase, leading to a wide range of α between the nitride and the β-matrix). The reaction product is zirconium nitride (ZrN); the reaction is exothermic, releasing approximately 82 kcal/mole. (The energy released in forming the oxide is approximately 262 kcal/mole.) The thickness of the zirconium nitride layer has been found[149] to be much smaller than that of the dissolution zone (in the temperature range 750°C- 1000°C) which indicates that the rate constant for film formation is considerably smaller than the rate constant for nitrogen dissolution. In fact, at 1000°C, 84% of the total nitrogen uptake was due to dissolution in the metal.

The role of nitrogen in the high temperature reaction of zirconium with air has been investigated[151]. The reaction process is multiphase in nature. Adjacent to the β-phase of the zirconium is a layer of α-phase (stabilized by both oxygen and nitrogen) and a surface layer of ZrO_2. In general, a certain amount of nitride (ZrN) is formed. For temperatures up to approximately 1050°C the nitride is found as a layer between the stabilized α-phase and the oxide layer; above 1050°C the nitride occurs as discrete particles dispersed in the oxide.

It is doubtful whether any appreciable amount of nitride is formed in the problem currently being considered. At the lower temperatures (during heat up) the reaction rate is very slow. Once rapid oxidation is initiated (approximately 900°C) the self-sustaining reaction proceeds very quickly, and

there may not be sufficient time for ZrN to be formed. Any nitride that does form, however, will contribute to the chemical energy release for the self-sustaining reaction.

The reaction rate of zirconium is higher with air than with oxygen alone. The explanation advanced is that nitrogen dissolves in ZrO_2. By replacing oxygen ions in the oxide structure, the higher valency nitrogen can increase the anion vacancy concentration, thus permitting a higher rate of diffusion of oxygen through the anion-deficient zirconia.

In sum, there are a number of uncertainties associated with the zircaloy-air reaction equation. These are particularly important above 900°C where rapid oxidation occurs. The most significant appear to be (i) the difference in the oxidation rates of zirconium and zircaloy, and (ii) the multiphase nature of the oxidation process itself at these temperatures. The results given above in Section C and D (i.e. for zirconium vs. zircaloy-4, and oxygen consumption rate vs. oxidation rate, respectively) apply to oxidation in steam only. Analogous results are expected for oxidation in air, i.e. it is expected that the oxidation rate in zircaloy will be greater than that in zirconium, and the rate of oxygen consumption will be greater than the rate of oxide formation in both materials. The relative magnitude of these effects cannot be deduced from the steam oxidation data. What are required are new experiments and computer modeling (similar to those carried out by Pawel and Campbell[136] for oxidation in steam) for the high temperature reaction of zirconium and zircaloy with air. In lieu of these, we suggest that additional calculations be performed for two other zirconium-air reaction correlations which will serve as bounds for those presented in Figure 1. (a) The high temperature correlation for zirconium (above the phase change of ZrO_2) should be multiplied by a factor m_1 to account for the higher reaction rate in zircaloy. (b) The correlations above the α-β transformation temperature should be divided by a factor m_2 to account for the difference in oxygen consumption rate and rate of oxide formation. Values of m_1 and m_2 as large as five should be considered.

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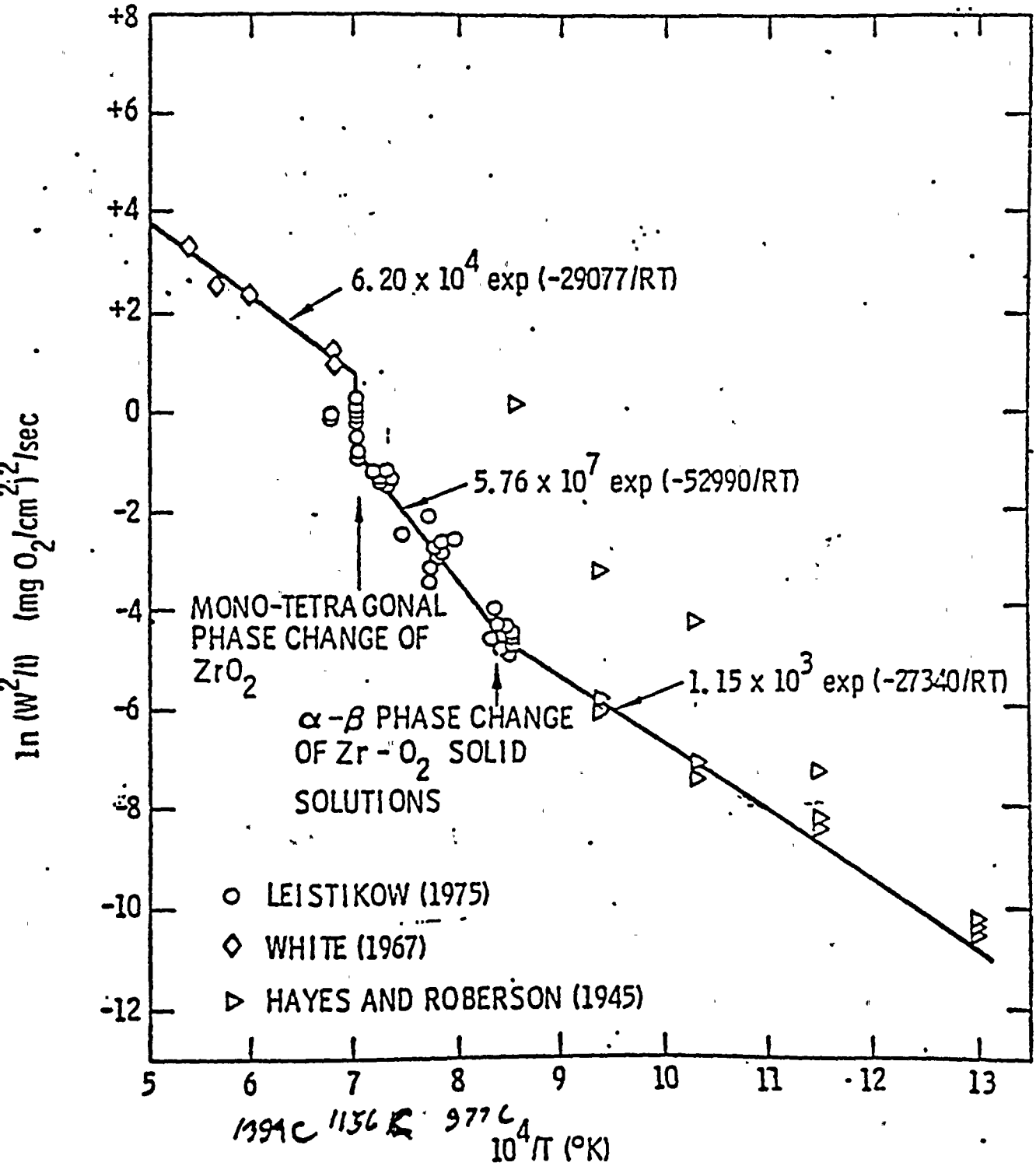


FIGURE 1 CORRELATIONS FOR ZIRCONIUM OXIDATION IN AIR (FROM REF. 1) ≠Zr-4

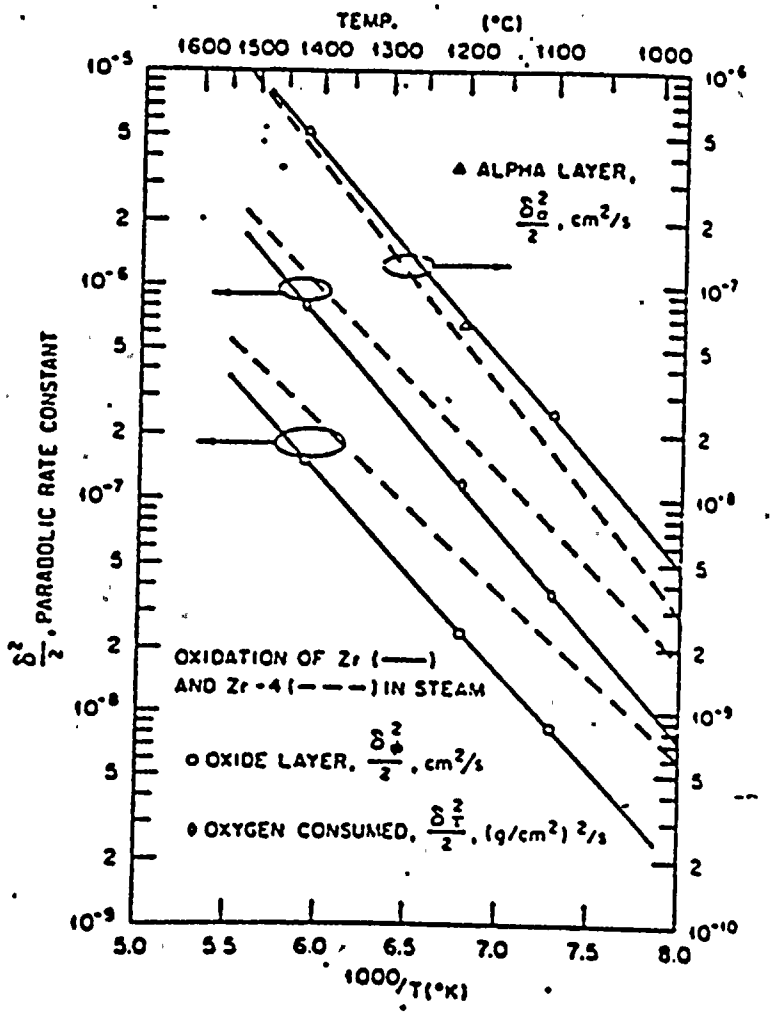


FIGURE 2 PARABOLIC RATE CONSTANTS FOR OXIDE LAYER GROWTH, α-LAYER GROWTH, AND OXYGEN CONSUMPTION FOR THE REACTION OF ZIRCONIUM (SOLID LINES) AND ZIRCALOY-4 (DASHED LINES) WITH STEAM. THE RATE CONSTANTS FOR OXYGEN CONSUMED (WEIGHT GAIN) WERE DETERMINED FROM MODELING ANALYSES (FROM REF. 136).

Attachment 2

BROOKHAVEN NATIONAL LABORATORY

MEMORANDUM

DATE: January 27, 1986

TO: Kenneth Perkins

FROM: John Weeks

SUBJECT: Parabolic Rate Constants for Oxidation of Zircaloy-4 in Dry Air

On March 27, 1985, Lou Teutonico (1) provided you with an assessment of the knowledge of oxidation kinetics of zircaloy-4 in air and steam based on the literature available at that time, as part of our overall assessment of the Sandia report, NUREG/CR/0649, under our NRC contract FIN A-3786. Among other things, Lou showed significant differneces in the oxidation kinetics of zirconium metal and zircaloy-4 in steam, as evidenced by the work of Pawel and Campbell of Oak Ridge National Laboratory (2). Except for this work, there were little available data at temperatures above 1100°C, where rapid reaction rates are expected, except for the 1967 data by White on unalloyed zirconium (3) which show significantly higher rates than would be expected by extrapolation of results obtained at lower temperatures.

When work was initiated on this program, NRC offered to obtain for us some more recent unpublished German data. These were never received through that source. After Teutonico left Brookhaven, I attempted to reevaluate the work he had done in the context of comparing the Pawel and Campbell data with the Sandia curve. It is immediately apparent that the Pawel and Campbell parabolic rate constants are considerably lower than the curve used by

Sandia. Figure 1 shows this comparison. Subsequently, while at the International Conference on Environmental Degradation of Nuclear Materials in September, I discussed the subject further with Dr. Hee Chung of Argonne National Laboratory and Dr. Friedrich Garzarolli of KWU, through whom I requested the unpublished German data. Chung pointed out that, while the rate controlling step in the high temperature oxidation of zirconium or zircaloys is the diffusion of oxygen through the oxide and/or through the solid solution of oxygen in zircaloy that underlies it in both steam and air oxidation, there is a significant decrease in the oxidation rate observed in a steam environment due to an effect of the hydrogen produced during this oxidation on these diffusion constants. He pointed out that, while this effect has been observed by several workers, it is not sufficiently quantified to permit us to use high temperature steam data (such as some of his own, those of Prater and Courtright at PNL (4) and those of Pawel and Campbell at Oak Ridge) to estimate oxidation rates under our fuel pool accident scenario. This leaves us, therefore, with only the White data in the high temperature range.

Garzarolli advised me that most of the German data were generated by Siegfried Leistikow at Karlsruhe. Following the conference, I wrote both to Garzarolli and Leistikow, and from both sources received copies of Leistikow's more recent data. In particular, Leistikow sent me, not only unpublished curves in air and steam for oxidation kinetics of zircaloy-4, but several internal reports, in German, that contain the results of a few short-term experiments above 1100°C. Appendix I to this memorandum gives the cover letter from Leistikow and his recent unpublished data. You will note from the letter that data at temperatures above 1100°C may be available in a year or

so. I also received the report KFK 2587 dated March 19, 1978, which shows some high temperature oxidation rate measurements on zircaloy-4 in air, oxygen, and steam. I've included this figure as Appendix II. The few data available above 1100°C show that the oxidation rates of zircaloy-4 are much greater in air than they are in either oxygen or steam. Leistikow's new data show roughly a parabolic corrosion rate behavior (slope of 1/2 on the log log plot) for the first 30-60 minutes in both air and steam. They also show that the difference between the air and the steam rates increases with temperature. After 30-60 minutes, however, the rate at all but the highest temperature increased dramatically, especially in air. This may be due either to difficulty in controlling the temperature of the highly exothermic zirconium/air oxidation, or to some "breakaway" type phenomenon in the surface oxides exposing the bare metal underneath. Leistikow drew his curves to suggest a new leveling off, at least at 950 and 1000°C after long times ($t > 90$ min.).

At lower temperatures, zirconium and zircaloy are know to oxidize according to the cubic law, which would mean a slope of 1/3 on a log log plot. The high temperature data used by Sandia were all approximated using parabolic growth, which is more typical of diffusion controlled phenomenan such as are believed to occur at high temperatures. The new German data show a slope somewhere between 1/3 and 1/2 for the first 30 minutes or so. In an attempt to compare these data with the Sandia curve, I drew lines with a slope of 1/2 through the data for the first 30 minutes in air, and from them calculated a parabolic rate constant which I have compared with the Sandia curves and the Pawel and Campbell data in steam in figure 1. I also used the

same approach on the much (approximately 10 x) higher long-term oxidation rates at 950, 1000, and 1100°C, on several of his curves in the first 60 minutes or so obtained in steam, and on the short-term data in Appendix II at 1160°C. These rate constants are also shown in figure 1. It is apparent, therefore, that the German steam data and those of Pawel and Campbell for zircaloy-4 in steam consistent in the temperture range in which they overlap. The new German air data are consistant with some of their own work (at short exposure times) published some years earlier (5). From the new German data, I suggest the rate equation:

$$\frac{(W)^2}{t} = 3.09 \times 10^8 \exp\left(-\frac{56{,}000}{RT}\right) \qquad (1)$$

where W is in mg O_2 reacted per square cm, t is in seconds, and T is in °K.

The instantaneous rate, $\frac{dw}{dt}$ at time t and temperature T is given by

$$\frac{dw}{dt} = \frac{1}{2w} 3.09 \times 10^8 \exp\left(-\frac{56{,}000}{RT}\right) \qquad (2)$$

The Sandia curve shows an abrupt increase in oxidation rate at $10^4/T = 7$, which they attribute to the mono-tetragonal phase change of ZrO_2. As can be seen in figure 1, the Pawel and Campbell data do not show such an abrupt change at this temperature; however, they were obtained in steam. The recent results of Prater and Courtright (4) (which were presented at the 1985 Symposium on Zirconium) show that for reactions in steam they find a similar jump at temperatures as high as 1500°C (1/T is 5.5×10^{-4}). This may be due to effects of the hydrogen produced by steam reaction on the oxide structure on the zircaloy. Unfortunately, Prater and Courtright plotted their data in

terms of thickness of the ZrO_2 film, and thus these could not readily be transferred to figure 1 which is in wt. of O_2 reacted. Since a considerable amount of the oxygen that reacts either from air or steam exists in high concentration solid solutions in the zircaloy, and since we are concerned in our accident scenario with the heat generated by this reaction, I think it is important that we consider the total oxygen consumed rather than just the thickness of the layer. I would anticipate the free energy of formation per gram atom of oxygen reacted be approximately the same for the zirconium oxygen solid solution as for ZrO_2 at these high temperatures. I have included Prater and Courtright's figure as Appendix III.

Conclusions

Based on the information available to date it appears impossible to justify any major changes to the Sandia equation; in particular, the curve from the work of White at temperatures above 1150°C appears to be all we have. However, this was obtained on unalloyed Zr, not zircaloy, and the higher rates for zircaloy-4 over those for unalloyed Zr observed by Pawel and Campbell in steam may also exist in air. For temperatures from 800-1150°C, I think the new German data fit in well with what was previously observed, and suggest using equation 1 given above. However, if the exposure is for periods greater than 30 minutes, this curve may not be conservative, as shown in the new German data plotted in figure 1.

cc: W.Y. Kato
W.T. Pratt
V.L. Sailor
L.J. Teutonico

1. L.J. Teutonico, Memo to K. Perkins regarding FIN A-3786, March 27, 1985.

2. R.E. Pawel and J.J. Campbell, Zirconium in the Nuclear Industry, ASTM STP 754, pg. 370, 1982.

3. J.H. White, GEMP-67, pg. 151, 1967.

4. J.T. Prater and E.L. Courtright, Oxidation of Zircaloy-4 in Steam at 1300 to 2400°C, Presented at Seventh International Conference on Zirconium in the Nuclear Industry, June 24-27, 1985.

5. S. Leistikow et al., KFK-2262, pg. 233, 1976.

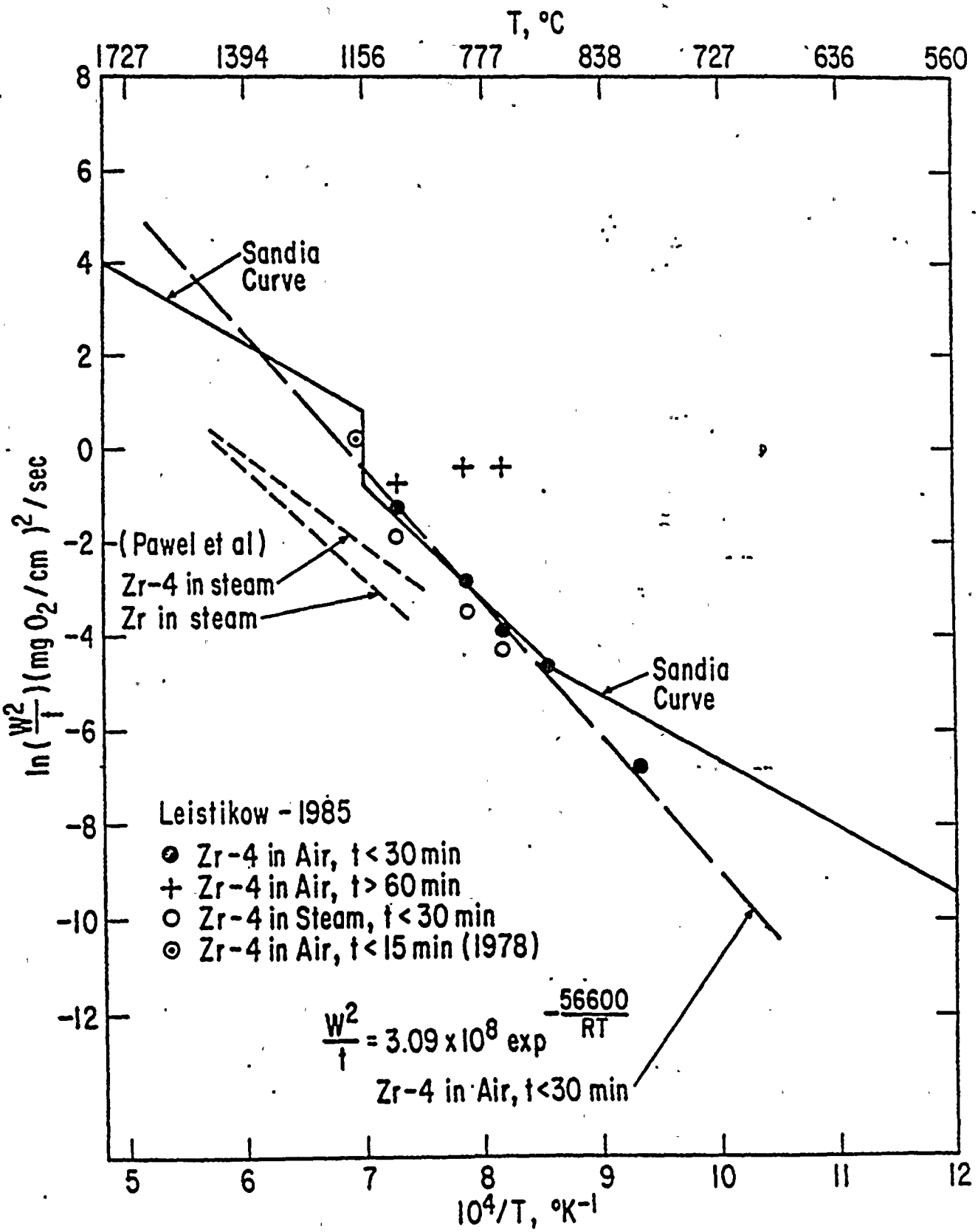

T, °C
1727
1394
1156
777
838
727
636
560
Sandia Curve
(Pawel et al)
Zr-4 in steam
Zr in steam
Sandia Curve
Leistikow - 1985
Zr-4 in Air, t < 30 min
Zr-4 in Air, t > 60 min
Zr-4 in Steam, t < 30 min
Zr-4 in Air, t < 15 min (1978)
$\frac{W^2}{t} = 3.09 \times 10^8 \exp -\frac{56600}{RT}$
Zr-4 in Air, t<30 min
$\ln(\frac{W^2}{t})(mg\ O_2/cm)^2/sec$
$10^4/T, °K^{-1}$

Figure 1

Appendix I

Kernforschungszentrum
Karlsruhe
Gesellschaft mit beschränkter Haftung

Kernforschungszentrum Karlsruhe GmbH Postfach 3640 D-7500 Karlsruhe 1

Institut für
Material- und Festkörperforschung II

Dr. John Weeks
Materials Technology Division
Brookhaven National Laboratory

Upton, Long Island, N.Y. 11973
USA

Datum: 21.10.1985 - Th.
Bearbeiter: Dr. S. Leistikow
Telefon: 07247/82 - 2915
Ihre Mitteilung:

Dear Dr. Weeks,

Dr. Garzarolli was right in telling you about our Zircaloy-4 oxidation experiments in air which in fact are not yet published. Sorry that we did not work above 1100°C which in fact could be done next year. So I send our curves

- Zircaloy-4 in air 800 - 1100°C
- Zircaloy-4 in steam and air 800-1100°C

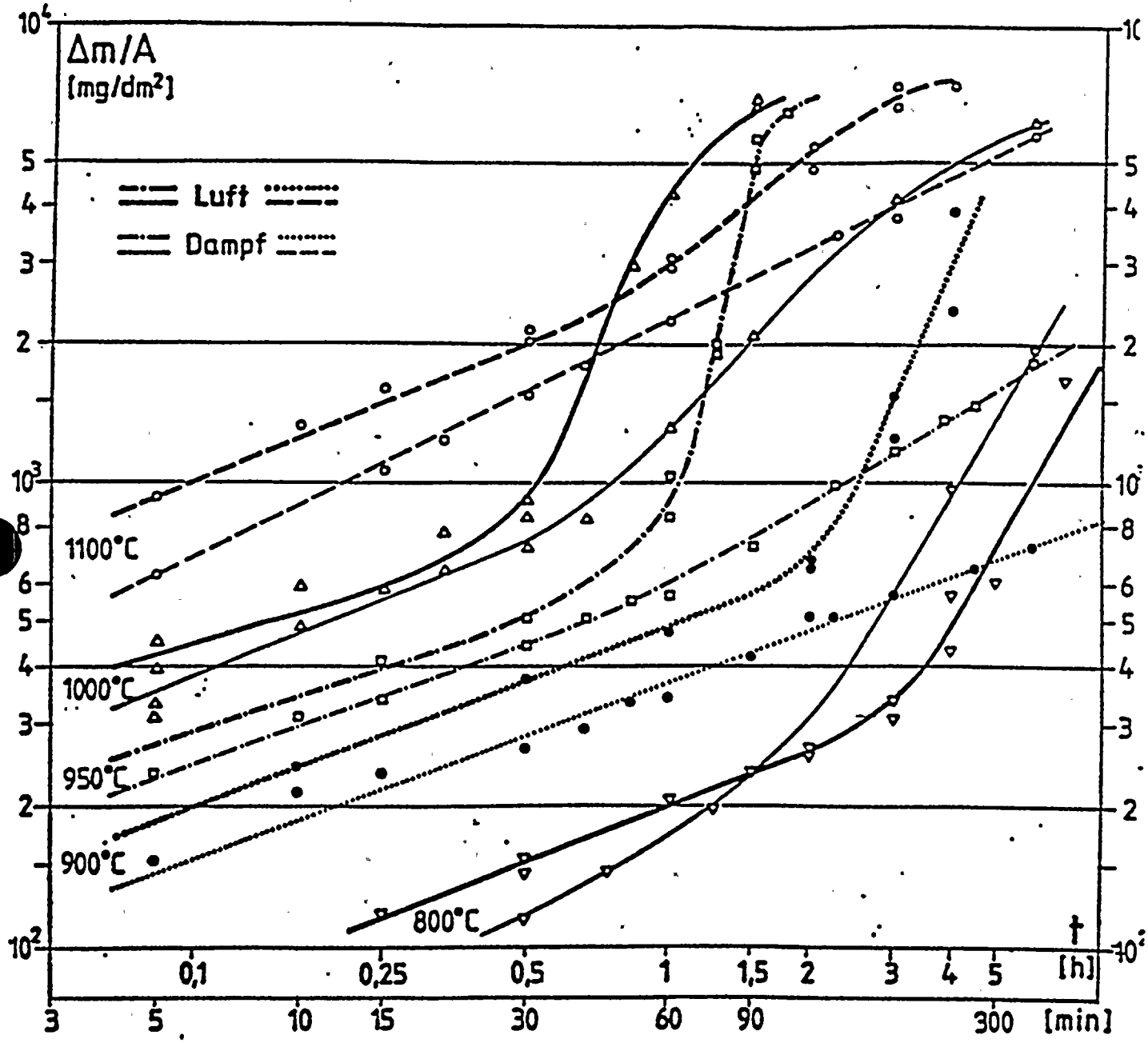
and add some other, more general publications we wrote on Zircaloy-4 behavior under LWR accident conditions.

I was aware of the problem (normal operating/accident conditions) when I applied for presentation of my paper to the Monterey Conference. Only your first positive reaction gave me hope. A participation after the rejection of the paper was then excluded because we are contributing to LWR problems only under the safety aspect - which in fact have their own series of conferences.

In case you have further questions don't hesitate to ask me. We dispose in case of the air oxidation experiments about a lot of other informations.

Very sincerely,

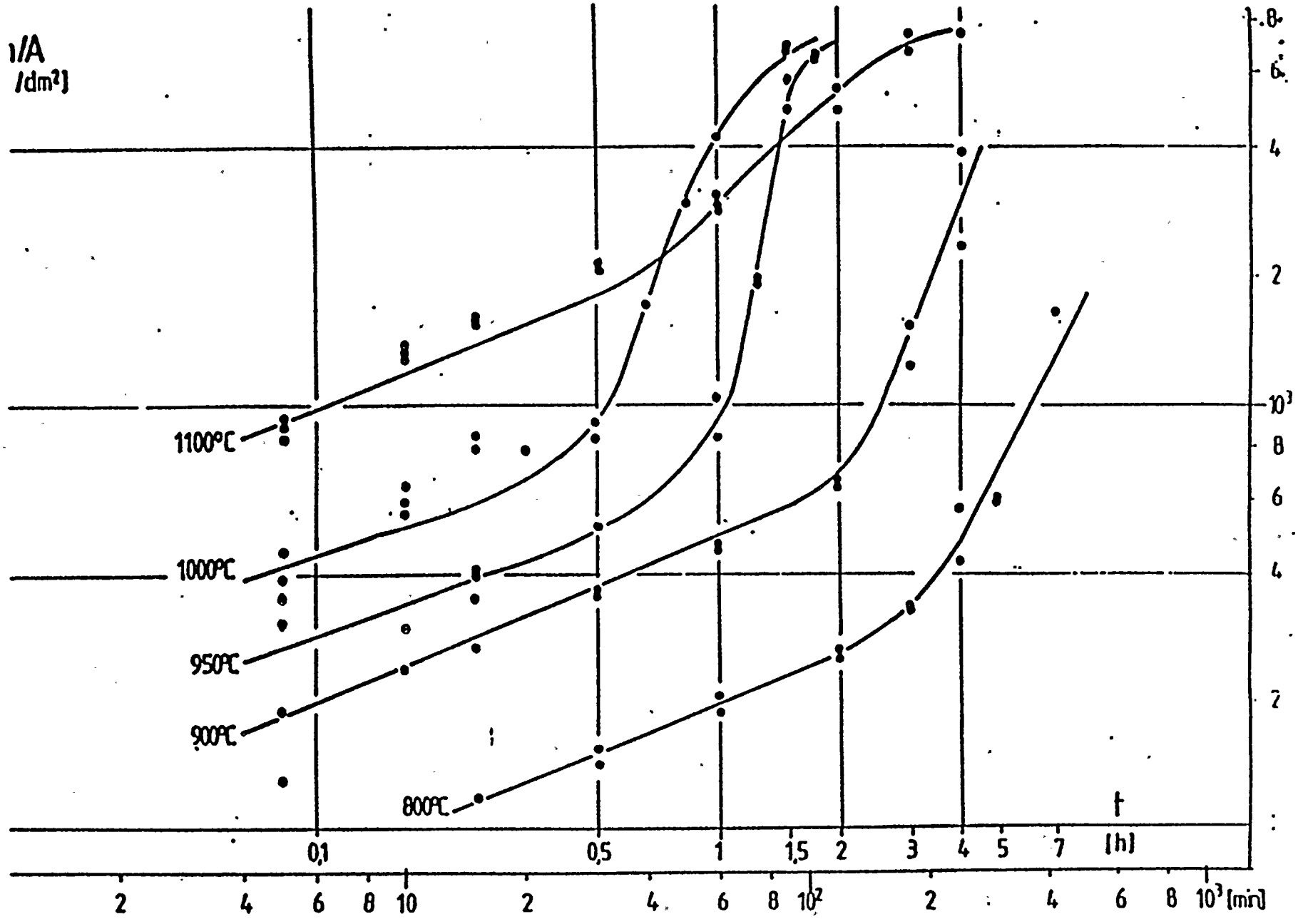
Siegfried Leistikow



KfK

Isotherme Zircaloy-4 / Oxidation

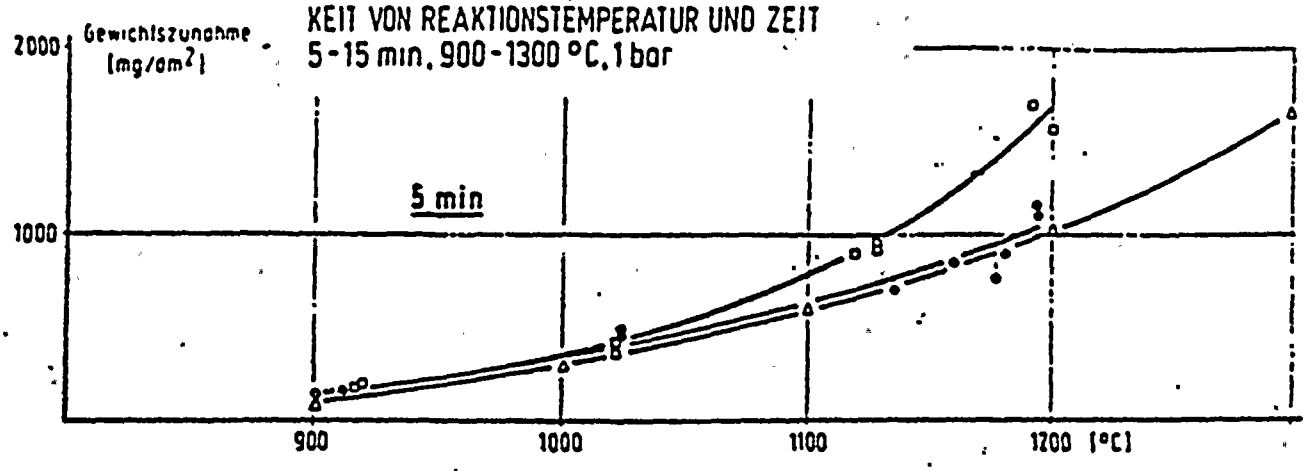
Vergleich der Massenzunahmen . 1800° 1100°C -7b1

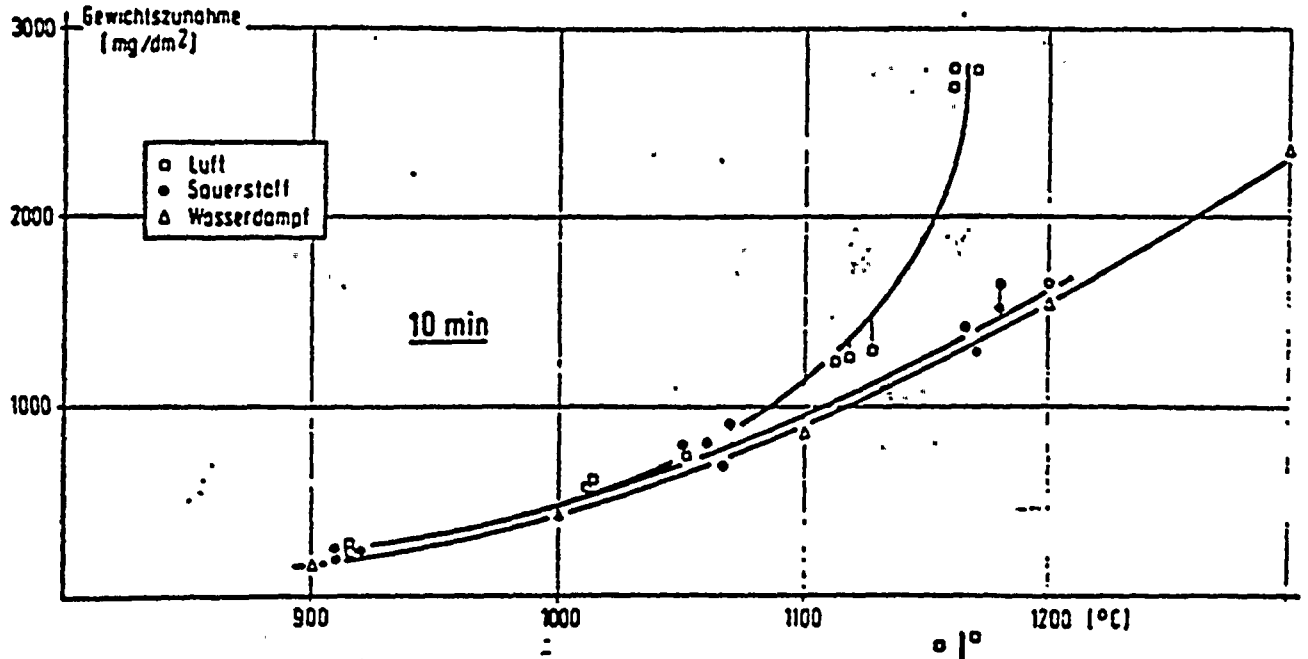


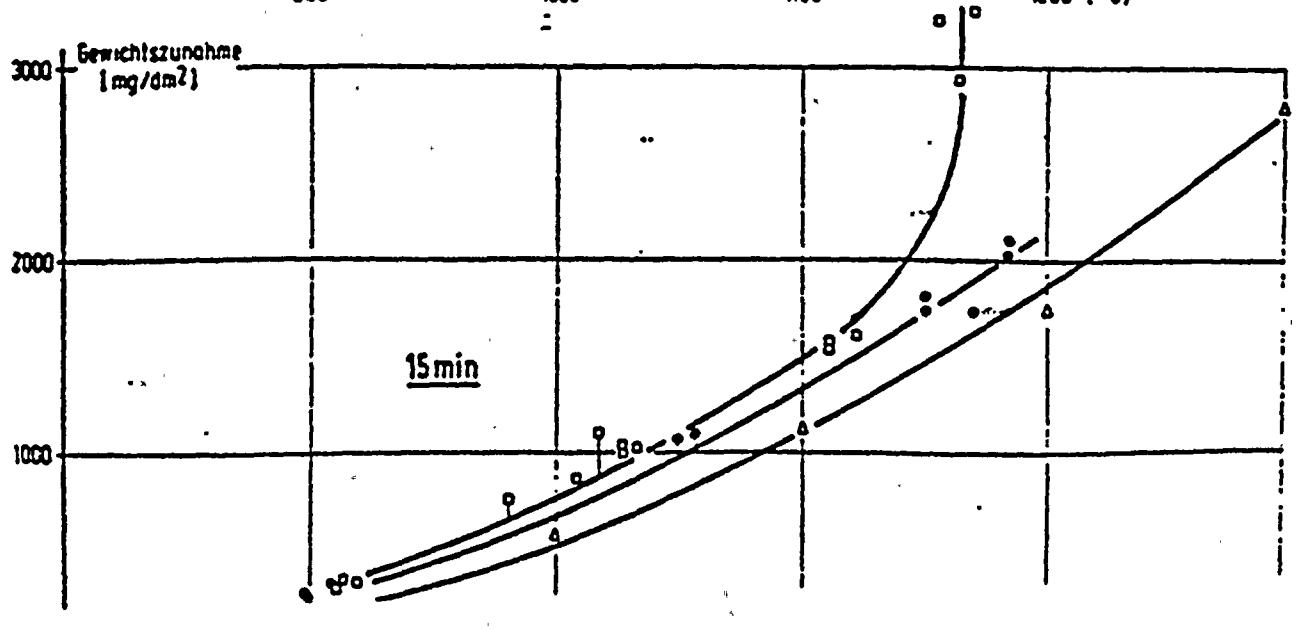
ien Zircaloy-4/Luft-Oxidation

ZIRCALOY 4 - HOCHTEMPERATUR-OXIDATION

DURCH WASSERDAMPF, SAUERSTOFF UND LUFT IN ABHANGIGKEIT VON REAKTIONSTEMPERATUR UND ZEIT
5 - 15 min, 900 - 1300 °C, 1 bar







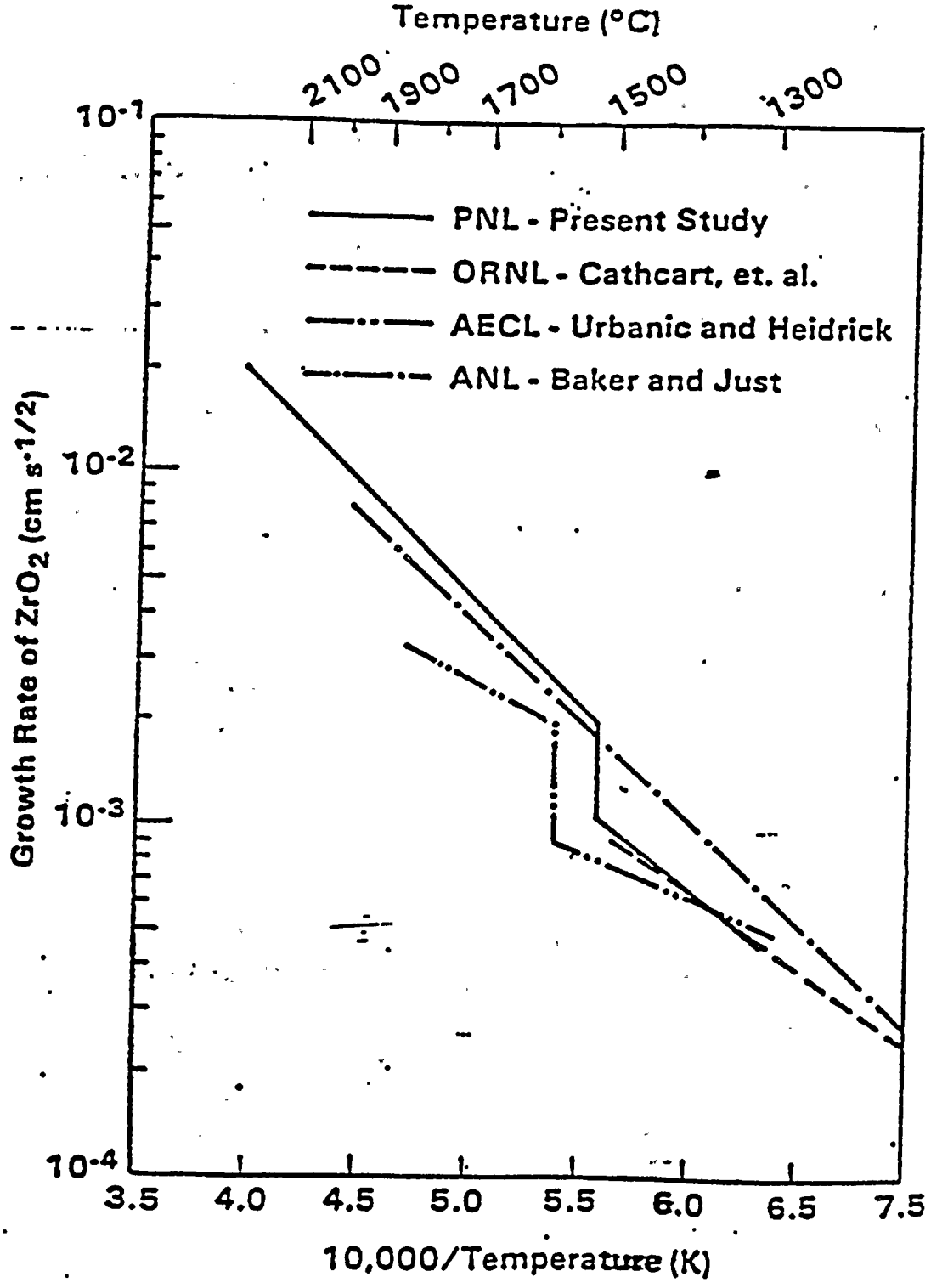

Temperature (°C)
2100
1900
1700
1500
1300
10-1
10-2
10-3
10-4
PNL - Present Study
ORNL - Cathcart, et. al.
AECL - Urbanic and Heidrick
ANL - Baker and Just
Growth Rate of ZrO2 (cm s-1/2)
3.5
4.0
4.5
5.0
5.5
6.0
6.5
7.5
10,000/Temperature (K)

Figure 5