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WASTES II Model Storage Requirements Benchmark Testing

**M. R. Shay
R. C. Walling
M. K. Altenhofen**

September 1986

**Prepared for the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830**

**Pacific Northwest Laboratory
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WASTES II MODEL STORAGE REQUIREMENTS
BENCHMARK TESTING

M. R. Shay
R. C. Walling
M. K. Altenhofen

September 1986

Prepared for
the U.S. Department of Energy
under Contract DE-AC06-76RL0 1830

Pacific Northwest Laboratory
Richland, Washington 99352

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SUMMARY

The Pacific Northwest Laboratory (PNL) has conducted a study to benchmark results obtained from using the Waste System Transportation and Economic Simulation - Version II (WASTES II) model (Shay and Buxbaum 1986) against information published in the Spent Fuel Storage Requirements report (DOE 1984). The WASTES model was developed by PNL for use in evaluating the spent-fuel storage and transportation requirements and costs for the U. S. Department of Energy (DOE). The Spent Fuel Storage Requirements report is issued annually by the DOE and provides both historical/projected spent fuel inventory data and storage requirements data based on information supplied directly from utilities. The objective of this study is to compare the total inventory and storage requirements documented in the Spent Fuel Storage Requirements report with similar data that results from use of the WASTES model.

Three differences have been identified as a result of benchmark testing. Two minor differences are present in the total inventory projected and the equivalent metric tons of uranium (MTU) of spent fuel requiring storage. These differences result from the way reinserted spent fuel is handled and the methods used to calculate mass equivalents. A third difference is found in the storage requirements for the case that uses intra-utility transshipment. This discrepancy is due to the Oyster Creek reactor, which is shown to not require additional storage in the Spent Fuel Storage Requirements report, even though there is no destination reactor of the same type within its utility. The discrepancy was corrected soon after the 1984 Spent Fuel Storage Requirements report was issued and does not appear in more recent documents (DOE 1985).

The results of benchmark testing demonstrate that the WASTES II model can be used to predict the requirements for additional at-reactor storage capacity. In addition, these results provide assurances that the WASTES II model produces reliable estimates of the spent-fuel logistics resulting from alternative spent-fuel management systems. It is recommended that the WASTES II model be benchmarked against other models of comparable design when they are available.

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1.0 INTRODUCTION

The WASTES II model was developed by PNL for use in evaluating the spent fuel storage and transportation requirements and costs for the DOE. As part of model development, benchmark tests were conducted to compare the data that resulted from using WASTES II with the information published in the 1984 Spent Fuel Storage Requirements report (DOE 1984). This report provides a brief description of the two independent approaches and discusses the results of the benchmark tests. The total inventories and requirements for additional at-reactor storage capacity are compared for the maximum pool capacity and intra-utility transshipment cases described in the Spent Fuel Storage Requirements report.

2.0 WASTES II MODEL DESCRIPTION

The WASTES II model simulates a user-defined system of nuclear spent-fuel generation, transportation, storage and final disposal. The number of each type of receiving facility, their locations and capacities, and priorities for waste acceptance are all specified by the user. The simulation is initiated with input data files that contain information about reactor-site characteristic and spent-fuel discharges. The reactor-site data file provides reactor site-specific data for each reactor present in the simulation. The reactor discharge data file provides the characteristics of each batch of spent fuel discharged in the simulation. From this point, the model is controlled by a user-specified combination of source- and destination-originated transfers.

During the simulation, spent-fuel transfers occur when a reactor pool violates the full core reserve (FCR) storage margin or when the reactor is decommissioned. At these times, the source reactor checks destination facilities to see if they can accept spent fuel. If the FCR margin cannot be maintained, the simulation will attempt to transship to another reactor pool, when transshipment is allowed. For the transshipment case, material will be transshipped to the storage pool with the largest capacity within the utility network. If transshipment is not allowed or if the transshipment is refused, the reactor will place excess material in dry storage. A dry storage facility is assumed to exist for each reactor and is allowed to grow as necessary to contain spent fuel that cannot be shipped to any other facility. In this way, the dry-storage inventories represent the requirements for additional at-reactor storage capacity.

3.0 SPENT FUEL STORAGE REQUIREMENTS REPORT DESCRIPTION

The Spent Fuel Storage Requirements report is published annually for the DOE Commercial Spent Fuel Management (CSFM) Program. The 1984 estimates of at-reactor storage requirements are based on information supplied by nuclear utilities during CY 1983. The Spent Fuel Storage Requirements report provides a basis for planning the DOE's Federal Interim Storage Program and various research and development activities within the CSFM Program. These estimates are based on the maximum at-reactor pool storage capacities of individual reactor pools, as determined by the utilities, and allowances for maintaining an FCR margin for each separate reactor. A single FCR margin is maintained for all units at multiple reactor stations sharing the same spent-fuel storage pool, or separate storage pools with interconnections for spent-fuel transfers. Two reference cases were used to develop the estimates. The first case assumes that transshipments occur only as currently planned by the utilities. The second case assumes that there are no constraints on transshipments of spent fuel among reactors of like type within a given utility system, thus delaying the utilities need for additional at-reactor storage capacity. Such transshipments are included in the spent-fuel management plans of several utilities. The two reference cases, therefore, define a range of potential storage requirements.

4.0 RESULTS

Cumulative total inventory and at-reactor storage requirements are shown in Table 4.1 for the maximum pool capacity case and Table 4.2 for the intra-utility transshipment case. For each case, there are minor differences in the total inventory amount for the period of time between 1983 and 1993. In addition, there are minor differences in the storage requirements expressed in terms of MTU's and a more significant difference in the storage requirements beginning in year 1992 for the case where transshipment occurs. These differences are explained below.

As shown in Table 4.1, the differences in total inventory amount are due to how reinserted spent fuel assemblies are accounted for with each approach. The Spent Fuel Storage Requirements report counts all fuel which has ever been discharged in the total inventory, including fuel which will be reinserted. Reinserted fuel is removed from inventory during the period of time that it is to be returned to the reactor, and is included again following final discharge. The WASTES II model is concerned with the logistics and costs associated with fuel which is to be stored or disposed, and the data used by WASTES only includes fuel which is permanently discharged. The inventory calculated by the WASTES model differs from that shown in the Storage Requirements report by less than 1 percent. This difference does not have a significant effect on the calculated storage requirements.

The differences in the at-reactor storage requirements expressed in MTU's are due to the different methods used to calculate equivalent mass equivalents. WASTES II bases the number of MTU's on individual spent-fuel discharge batch characteristics. WASTES II also maintains the age, exposure and heat generation rate characteristics of each spent-fuel batch. The approach used in the Spent Fuel Storage Requirements report uses the overall average assembly weight values specified for each reactor. As a result, WASTES II provides better estimates for MTU equivalents of spent fuel requiring storage. This difference is not judged to be significant since requirements are more often quoted on an assembly basis.

TABLE 4.1. WASTES II Model Benchmark Test Results--Maximum Pool Capacity

Year	WASTES II Model Results				Results from the Spent Fuel Storage Requirements Report			
	Storage Requirement		Total Inventory		Storage Requirement		Total Inventory	
	Assemblies	MTU	Assemblies	MTU	Assemblies	MTU	Assemblies	MTU
1983	0	0.0	37,113	10,074.3	0	0.0	37,271	10,142.0
1984	0	0.0	41,781	11,397.8	0	0.0	41,843	11,425.0
1985	3	1.2	47,628	13,143.4	3	1.0	47,672	13,163.0
1986	69	27.5	55,049	15,172.3	69	28.0	55,082	15,187.0
1987	363	117.3	62,984	17,449.1	363	119.0	63,010	17,461.0
1988	704	263.0	72,594	20,178.6	704	266.0	72,612	20,187.0
1989	1,268	460.6	81,036	22,647.2	1,268	459.0	81,053	22,655.0
1990	2,824	896.9	90,622	25,377.0	2,824	893.0	90,630	25,381.0
1991	5,032	1,441.7	100,703	28,199.4	5,032	1,429.0	100,707	28,201.0
1992	6,507	1,880.1	110,173	30,976.2	6,507	1,861.0	110,177	30,978.0
1993	8,874	2,500.8	120,368	33,876.8	8,874	2,472.0	120,368	33,877.0

As shown in Table 4.2, the difference in the storage requirements for the case with intra-utility transshipment beginning in year 1992 is due to the situation at Oyster Creek reactor. As shown in Appendix A, Oyster Creek is not affected by intra-utility transshipment because a destination reactor of the same type does not exist within its utility system. The Spent Fuel Storage Requirements report indicates that Oyster Creek reactor storage requirements are impacted by intra-utility transshipment. This discrepancy was examined and it was found that the results reported in the Spent Fuel Requirements report were in error, but had been corrected prior to this benchmark. Comparison of WASTES II model results with more recent reports (Heeb, Libby, and Holter 1985, DOE 1985) indicate full agreement.

TABLE 4.2. WASTES II Model Benchmark Test Results--Intra-Utility Transshipment

Year	WASTES II Model Results				Results from the Spent Fuel Storage Requirements Report			
	Storage Requirement		Total Inventory		Storage Requirement		Total Inventory	
	Assemblies	MTU	Assemblies	MTU	Assemblies	MTU	Assemblies	MTU
1983	0	0.0	37,113	10,074.3	0	0.0	37,271	10,142.0
1984	0	0.0	41,781	11,397.8	0	0.0	41,843	11,425.0
1985	0	0.0	47,628	13,143.4	0	0.0	47,672	13,163.0
1986	0	0.0	55,049	15,172.3	0	0.0	55,082	15,187.0
1987	132	24.4	62,984	17,449.1	132	23.0	63,010	17,461.0
1988	132	24.4	72,594	20,178.6	132	23.0	72,612	20,187.0
1989	337	65.6	81,036	22,647.2	337	61.0	81,053	22,655.0
1990	663	152.8	90,622	25,377.0	663	147.0	90,630	25,381.0
1991	1,575	382.0	100,703	28,199.4	1,575	369.0	100,707	28,201.0
1992	2,756	690.1	110,173	30,976.2	2,620	640.0	110,177	30,978.0
1993	4,045	1,084.6	120,368	33,876.8	3,909	1023.0	120,368	33,877.0

5.0 CONCLUSIONS

The results of benchmark testing demonstrate that the WASTES II model accurately predicts the requirements for additional at-reactor storage capacity for a no-acceptance scenario. In addition, these results provide assurances that the WASTES II model produces reliable estimates of the storage requirements resulting from alternative spent-fuel management systems. It is recommended that that WASTES II be benchmarked against other models of comparable design when they are available.

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APPENDIX A

WASTES II MODEL RESULTS--ANNUAL STORAGE REQUIREMENTS

TABLE A.1. WASTES II Model Results--Annual Storage Requirements--Maximum Pool Capacity Case

RXID	Reactor Name	Number of Assemblies									Metric Tons of Uranium								
		1985	1986	1987	1988	1989	1990	1991	1992	1993	1985	1986	1987	1988	1989	1990	1991	1992	1993
8	ARKANSAS NUCL ONE-1	0	0	0	0	0	0	0	0	1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.5
10	CALVERT CLIFFS-1	0	0	0	0	0	0	0	45	72	0.0	0.0	0.0	0.0	0.0	0.0	0.0	16.8	24.5
12	PILGRIM-1	0	0	0	0	0	0	32	0	192	0.0	0.0	0.0	0.0	0.0	0.0	6.2	0.0	37.1
14	ROBINSON-2	0	0	0	0	0	43	52	52	52	0.0	0.0	0.0	0.0	0.0	18.8	22.3	22.3	22.3
15	BRUNSWICK-2	0	0	0	0	0	179	0	180	180	0.0	0.0	0.0	0.0	0.0	33.6	0.0	33.7	33.7
16	BRUNSWICK-1	0	0	0	0	0	0	179	180	0	0.0	0.0	0.0	0.0	0.0	0.0	33.4	33.7	0.0
32	LA SALLE-1	0	0	0	0	0	328	464	0	448	0.0	0.0	0.0	0.0	0.0	60.1	85.0	0.0	82.1
40	INDIAN POINT-2	0	0	0	0	0	0	0	0	21	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	9.6
42	PALISADES	0	0	19	68	0	68	68	0	68	0.0	0.0	7.8	27.8	0.0	27.8	27.7	0.0	27.1
49	OCONEE-1	0	0	0	0	0	64	64	68	132	0.0	0.0	0.0	0.0	0.0	29.7	29.9	31.6	61.4
51	OCONEE-3	0	0	12	0	68	68	0	68	0	0.0	0.0	5.6	0.0	31.8	31.6	0.0	31.6	0.0
66	TURKEY POINT-3	0	9	0	68	52	0	68	52	0	0.0	4.1	0.0	30.5	23.0	0.0	30.9	23.6	0.0
67	TURKEY POINT-4	0	0	31	64	0	64	64	0	64	0.0	0.0	14.0	28.9	0.0	28.9	28.8	0.0	29.6
68	ST. LUCIE-1	0	0	21	0	88	76	0	88	76	0.0	0.0	8.3	0.0	33.7	28.8	0.0	33.5	29.6
85	OYSTER CREEK	0	0	0	0	0	0	0	136	0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	26.4	0.0
99	COOPER	0	0	0	0	0	0	0	0	82	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	16.0
106	MILLSTONE-1	0	0	132	0	200	0	200	0	200	0.0	0.0	24.4	0.0	39.1	0.0	37.9	0.0	38.3
107	MILLSTONE-2	3	57	73	81	0	81	77	73	0	1.2	22.3	26.9	31.3	0.0	31.7	30.3	29.3	0.0
112	MONTICELLO	0	0	0	0	0	3	104	104	0	0.0	0.0	0.0	0.0	0.0	0.6	20.1	20.1	0.0
113	PRAIRIE ISLAND-1	0	0	0	0	5	80	80	80	80	0.0	0.0	0.0	0.0	2.0	32.0	32.0	32.0	32.0
124	PEACH BOTTOM-2	0	0	0	0	0	206	256	0	256	0.0	0.0	0.0	0.0	0.0	37.5	46.6	0.0	46.6
125	PEACH BOTTOM-3	0	0	0	0	0	196	256	0	256	0.0	0.0	0.0	0.0	0.0	35.7	46.6	0.0	46.6
133	FITZPATRICK	0	0	0	0	0	0	132	200	0	0.0	0.0	0.0	0.0	0.0	0.0	25.8	37.5	0.0
153	GINNA	0	0	0	0	0	26	28	28	28	0.0	0.0	0.0	0.0	0.0	10.0	10.9	11.0	10.9
167	SEQUOYAH-1	0	0	0	0	0	0	0	0	67	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	30.5
183	DAVIS-BESSE-1	0	0	0	0	0	0	0	0	31	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	14.6
189	SURRY-1	0	0	6	60	151	61	60	121	61	0.0	0.0	2.7	27.2	67.9	27.9	27.4	55.3	27.8

TABLE A.2. WASTES II Model Results--Annual Storage Requirements--Intra-Utility Transshipment Case

RXID	Reactor Name	Number of Assemblies							Metric Tons of Uranium						
		1987	1988	1989	1990	1991	1992	1993	1987	1988	1989	1990	1991	1992	1993
10	CALVERT CLIFFS-1	0	0	0	0	0	45	72	0.0	0.0	0.0	0.0	0.0	16.8	24.5
12	PILGRIM-1	0	0	0	0	32	0	192	0.0	0.0	0.0	0.0	6.2	0.0	37.1
15	BRUNSWICK-2	0	0	0	178	0	180	180	0.0	0.0	0.0	33.4	0.0	33.7	33.7
16	BRUNSWICK-1	0	0	0	0	180	180	0	0.0	0.0	0.0	0.0	33.6	33.7	0.0
40	INDIAN POINT-2	0	0	0	0	0	0	21	0.0	0.0	0.0	0.0	0.0	0.0	9.6
45	LA CROSSE	0	0	0	13	24	0	0	0.0	0.0	0.0	1.6	2.9	0.0	0.0
66	TURKEY POINT-3	0	0	0	0	68	52	0	0.0	0.0	0.0	0.0	30.9	23.6	0.0
67	TURKEY POINT-4	0	0	0	0	64	0	64	0.0	0.0	0.0	0.0	28.8	0.0	29.6
68	ST. LUCIE-1	0	0	0	26	0	88	76	0.0	0.0	0.0	9.7	0.0	33.5	29.6
69	ST. LUCIE-2	0	0	0	0	0	88	76	0.0	0.0	0.0	0.0	0.0	39.8	33.8
85	OYSTER CREEK	0	0	0	0	0	136	0	0.0	0.0	0.0	0.0	0.0	26.4	0.0
99	COOPER	0	0	0	0	0	0	82	0.0	0.0	0.0	0.0	0.0	0.0	16.0
106	MILLSTONE-1	132	0	200	0	200	0	200	24.4	0.0	39.1	0.0	37.9	0.0	38.3
112	MONTICELLO	0	0	0	3	104	104	0	0.0	0.0	0.0	0.6	20.1	20.1	0.0
113	PRAIRIE ISLAND-1	0	0	5	80	80	80	80	0.0	0.0	2.0	32.0	32.0	32.0	32.0
133	FITZPATRICK	0	0	0	0	132	200	0	0.0	0.0	0.0	0.0	25.8	37.5	0.0
153	GINNA	0	0	0	26	28	28	28	0.0	0.0	0.0	10.0	10.9	11.0	10.9
183	DAVIS-BESSE-1	0	0	0	0	0	0	31	0.0	0.0	0.0	0.0	0.0	0.0	14.6
189	SURRY-1	0	0	0	0	0	0	61	0.0	0.0	0.0	0.0	0.0	0.0	27.9
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