HOLTEC
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UPDATED THERMAL-HYDRAULIC ANALYSIS

of

SPENT NUCLEAR FUEL POOL

DONALD C. COOK NUCLEAR PLANT
INDIANA MICHIGAN POWER COMPANY

by

HOLTEC INTERNATIONAL

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SUMMARY OF REVISIONS LOG

HOLTEC REPORT HI-951389

Revision 1 contains several editorial changes and clarifications throughout the text.	
REVISION 1 CONTAINS THE FOLLOWING PAGES:	
Title Page	1
Review and Certification Log	1
Summary of Revisions Log	2
Table of Contents	1
Section 1	2
Section 2	1
Section 3	2
Section 4	3
Section 5	1
Section 6	1
Section 7	3
Tables	5
Figures	10

REVISION 0 CONTAINS THE FOLLOWING PAGES:	
Title Page	1
Review and Certification Log	1
Summary of Revisions Log	1
Table of Contents	1
Section 1	1
Section 2	1
Section 3	2
Section 4	3
Section 5	1

Section 6	1
Section 7	3
Tables	5
Figures	10

TABLE OF CONTENTS

1.0 INTRODUCTION

The Donald C. Cook Nuclear Plant, operated by American Electric Power (AEP) for Indiana Michigan Power, is a dual unit pressurized water reactor (PWR) generating station. The rated thermal power of the Unit 1 reactor is 3250 MW(t). The rated thermal power of the Unit 2 reactor is 3588 MW(t). Spent nuclear fuel (SNF) assemblies discharged from both reactors are stored in a common spent fuel pool (SFP). The SFP contains storage racks which provide sufficient storage locations for 3613 SNF assemblies. Heat is generated by the continuing radioactive decay of the stored SNF assemblies.

Recent modification to the SNF discharge schedules for the two reactors will alter the decay heat generated by SNF assemblies in the SFP. The new discharge schedules call for increasing both the cycle length and the number of assemblies discharged per cycle. While the longer cycle length increases the fuel decay time between discharges, reducing the decay heat contribution of previously discharged fuel, it also increases the reactor "operating" time of freshly discharged fuel, which increases the decay heat contribution of the freshly discharged assemblies. The purpose of this report is to characterize the thermal-hydraulic response of the SFP in light of the new discharge schedule. For all scenarios in this evaluation the in-core decay time (also referred to as the reactor hold time) for fuel from both reactors is assumed to be 100 hours.

The analyses documented herein parallel those contained in Holtec report HI-941183 [1]. The results contained in this report supersede the results of the earlier report. Previous Holtec

analyses of the Donald C. Cook SFP determined the fuel decay heat using the methods outlined in USNRC Branch Technical Position ASB 9-2 [2]. This analysis uses the ORIGEN-2 computer code [3] to determine the decay heat. The ORIGEN-2 decay heat calculations are more rigorous and more accurate than the approximate correlations of ASB 9-2.

2.0 ANALYSIS PROCEDURES

The following analyses are performed in the course of this thermal-hydraulic evaluation:

(i) Long-Term Decay Heat Calculation. This analysis is performed to calculate the accumulated decay heat of all previously discharged SNF assemblies in the SFP.

(ii) SFP Transient Thermal Response Determination. This analysis is performed to determine the SFP bulk temperature and decay heat profiles for the postulated final discharge into the pool.

(iii) Time-to-Boil Calculation. This analysis is performed to calculate the time required before boiling, in the wake of a postulated total loss of forced cooling of the SFP.

(v) Maximum Local Temperature Calculations. The maximum local water temperature and the maximum local fuel cladding temperature are determined, both with and without partial cell blockage, to evaluate the possibility of nucleate boiling on the surface of the fuel assemblies.

3.0 DISCHARGE SCHEDULES AND SCENARIOS

The fuel discharge schedules (including both historic and future fuel discharges) for both the Unit 1 and Unit 2 reactors are presented in Table 1 and Table 2, respectively. For postulated future fuel discharges, the fuel enrichments for Unit 1 and Unit 2 are assumed to be 3.5 % and 4.0 %, respectively. These enrichments are expected to provide a lower bound for future fuel enrichments, and therefore yield an upper bound for thermal power output. Because the analyses are performed for the last discharge scenario in which the SFP storage capacity is not exceeded, these two discharge schedules only include discharges up to the point where the SFP becomes full (cycle 25A) and not to the end-of-licensed reactor life.

A total of five discharge scenarios are evaluated in these analyses. These discharge scenarios are identical to those presented in previous licensing submittals. These five scenarios are summarized as follows:

Case 1A: Normal Discharge, 1 Cooling Train @ design flow

During the cycle 25A discharge from the Unit 1 reactor, a total of 84 fuel assemblies are placed in the SFP. All future fuel assemblies discharged are assumed to have been exposed to the maximum burnup of 52,200 MWD/MTU. The SFP water flow rate through the SFP cooling system (SFPCS) is assumed to be the design point value of 2,300 gal/min.

Case 1B: Normal Discharge, 1 Cooling Train @ maximum (as measured) flow

This case is identical to Case 1A, except that the SFP water flow rate through the SFPCS is assumed to be the maximum (as measured) value of 2,800 gal/min.

Case 2: Normal Discharge, 2 Cooling Trains

This case is identical to Case 1A, except that two cooling trains are operating.

Case 3: Back-to-Back Full Core Discharge, 2 Cooling Trains

The Unit 2 reactor has an unplanned shutdown 30 days after the cycle 25A shutdown of the Unit 1 reactor. A full core of 193 assemblies is removed from the Unit 2 reactor and placed in the fuel pool. There are two cooling trains operating at design flow rates.

The fuel assemblies from the Unit 2 reactor are discharged in three groups. The first group contains 65 assemblies with 64,800 MWD/MTU burnup, the second group contains 64 assemblies with 43,200 MWD/MTU burnup, and the third group contains 64 assemblies with 21,600 MWD/MTU.

While the fuel pool will not actually have enough storage locations to hold a full core offload after the normal cycle 25A discharge, this hypothetical case does provide a decay heat load that is guaranteed to bound any actual scenario (3800 assemblies).

Case 4: Back-to-Back Full Core Discharge, Single Cooling Train

This case is identical to Case 3, except that only one cooling train is operational. This case is not a design basis scenario for either the Cook Nuclear Plant or USNRC guidelines [4], and is presented for reference only.

4.0 ANALYSIS RESULTS

4.1 Long-Term Decay Heat Calculation

The decay heat contribution of all SNF in the SFP as of June 24, 2010 is determined using the proprietary Holtec computer program LONGOR [5]. Based on the discharge schedules in Table 1 and Table 2, the long-term decay heat is 14,117,944 Btu/hr. All subsequent analyses are performed with the long-term decay heat contribution held constant at this value.

4.2 SFP Transient Thermal Response Determination

The thermal response of the SFP and SFPCS is determined using the proprietary Holtec computer program BULKTEM [6]. The temperature and heat load profiles for each discharge scenario are presented graphically in Figures 1 through 10. Numerical results are summarized in Table 3.

The initial drop in temperature shown in Figures 1 through 5 is caused by an elevated initial temperature condition. The elevated initial temperatures have no effect on the analyses because all scenarios reach a steady-state temperature before commencement of fuel transfer.

The analysis shows that the limiting normal discharge scenario is Case 1A. The results of the analysis for this scenario demonstrates that the SFPCS can provides sufficient cooling to

maintain the temperature of the SFP below 155 °F. The coincident decay heat load (excluding evaporative heat losses) is 27.2 MBtu/hr. If two SFPCS trains are available, the peak temperature and coincident decay heat load become 129 °F and 29.3 MBtu/hr.

For a back-to-back full core discharge scenario (Case 3) as prescribed in SRP 9.1.3 [4], the maximum temperature is less than 147 °F and the coincident decay heat load is 54.3 MBtu/hr. In accordance with SRP 9.1.3, no single active failure need be associated with the full core discharge scenario. For reference, however, during a back-to-back full core discharge with only 1 SFP train operating (Case 4) the maximum temperature is maintained below 181 °F and the coincident heat load is 47.3 MBtu/hr.

4.3 Time-to-Boil Calculation

For each discharge scenario, the effects of a total loss of forced cooling is evaluated using the proprietary Holtec computer program TBOIL [7]. For each scenario, it is conservatively assumed that the loss of cooling occurs at the instant of peak bulk temperature and that no makeup water is available to the SFP. The evaluation results are summarized in Table 4. For the design basis cases, the minimum time between SFPCS failure and pool boiling is 6.08 hours (Case 3). The maximum boiloff rate for this scenario is 103.68 gal/min. For the design basis normal discharge (Case 1A), the time between SFPCS failure and pool boiling is 9.45 hours and the maximum boiloff rate is 63.35 gal/min.

4.4 Maximum Local Temperature Calculations

The maximum local water temperature and maximum local fuel cladding temperature are determined for the point in time where the bulk SFP temperature reaches its maximum value using the proprietary Holtec computer program THERPOOL [8]. Both unblocked and 50% blocked scenarios are evaluated. The results of this analysis are summarized in Table 5. For the limiting discharge scenario (Case 1), the maximum local water temperature is calculated as 163.6 °F and the local maximum fuel cladding temperature is 214.5 °F. If the limiting rack cells become blocked by 50%, the maximum water temperature increases to 223.5 °F and the maximum fuel cladding temperature increases to 254.3 °F.

The local boiling point is dependent on the local pressure. At a water depth of 23 feet, the local boiling point is 238 °F. The maximum local water temperature (50% blockage, above) is only 223.5 °F. Thus, nucleate boiling will not occur in the SFP, even under conditions of maximum heat flux and maximum bulk temperature.

5.0 CONCLUSION

The results presented in this report demonstrate that the maximum bulk spent fuel pool water temperature is conservatively bounded by a temperature of 155 °F, for normal discharge scenarios. This value is 6 °F less than that calculated in the previous evaluation [1]. Therefore, the modification to the fuel assembly discharge schedules has the effect of increasing the margin of safety over that established in the previous submittal.

6.0 REFERENCES

[1] Holtec Report HI-941183, "Spent Nuclear Fuel Pool Thermal-Hydraulic Analysis Report for Donald C. Cook Nuclear Plant", Rev. 2

[2] USNRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling", Rev. 2, 7/81

[3] A.G. Croff, "ORNL Isotope Generation and Depletion, A User's Manual for the ORIGEN-2 Computer Code", ORNL/TM-7175, RSIC/CCC-371, Oak Ridge National Laboratory, July, 1980.

[4] NUREG-0800, Standard Review Plan, Section 9.1.3

[5] Holtec Report HI-951390, "QA Documentation for LONGOR", Rev. 0

[6] Holtec Report HI-951391, "QA Documentation for BULKTEM", Rev. 0.

[7] Holtec Report HI-92832, "QA Documentation for TBOIL", Rev. 2.

[8] Holtec Report HI-92833, "QA Documentation for THERPOOL", Rev. 0.

7.0 ANSWERS TO RAI BY NRR

This section contains responses to questions posed by the USNRC in response to the previously submitted Holtec Report HI-941183. Any reference to tables or figures in this section refer to the corresponding articles in either the original licensing report (dated July, 26 1991) or Holtec Report HI-941183.

Q 1. It is assumed that the combined SFP Hx heat load and evaporative heat losses (as shown in Table 2.2 of Holtec Report HI-941183) are equivalent to the total decay heat generation in each case, e.g., Case 1A SFP Hx load 30.84 E6 BTU/Hr. + evaporative losses 3.14 E6 BTU/Hr., for a total of 33.98 E6 BTU/Hr. If this is incorrect, explain what the correct decay heat load is in each case and justify any differences.

A 1. Your assumption is correct as stated. The total decay heat generation is equal to the sum of the SFP Hx heat load and the evaporative cooling loss.

Q 2. A preliminary comparison was made of the decay heat generated by the 80 fuel elements deposited in the spent fuel pool in cases 1A, 1B and 2 for the decay times shown in Table 2.2 with similar cases in Table 5.5.1 of your previous submittal dated July 26, 1991 wherein the fuel was permitted to decay for 168 in lieu of the decay period of 100 hours presently requested. That comparison shows differences of 2.6 to 2.7 E6 BTU/Hr. in lieu of the differences you show of 0.71 to 0.86 E6 BTU/Hr. Justify your calculations.

A 2. The reduction in the reactor hold time will increase the decay heat of the freshly discharged fuel assemblies only, the decay heat from previously discharged fuels is not affected by the change in hold time. However, the fresh fuel decay heat accounts for less than 50% of the total heat generation. Additionally, Holtec Report HI-941183 incorporates changes to the refueling schedule which reduce the decay heat contribution of the previously discharged fuel. The reduction in the decay heat of the previously discharged fuel serves to limit the increase in the total decay heat generation rate.

Q 3. Explain whether you have deviated from Table 2.1 of Holtec report HI-941183 in using the number of those discharged assemblies and dates of discharges in calculating the heat generation of the spent fuel assemblies stored in the spent fuel pool. For example, you

stated that you calculated the heat generation for 80 fuel assemblies in a normal discharge batch in lieu of 76 shown in Table 2.1.

A 3. The decay heat calculations in Holtec Report HI-941183 are devised to provide an upper bound to any actual discharge scenarios. The calculation of the decay heat generation from all previously discharged fuels is based on the refueling schedule of Table 2.1. However, the normal discharge batch from Unit 1 contains more assemblies than does Unit 2. To provide an analysis that bounds all normal discharge scenarios, the final discharge batch size was assumed to be the Unit 1 batch size of 80 assemblies. This assumption serves to increase the conservatism of the analysis.

Q 4. Provide the decay heat generation rate for the assemblies deposited in the pool for each discharge cycle used in your calculations for Cases 3 and 4. If you do not use the discharges and cycle EFPD shown in Table 2.1 explain the method used and justify its application.

A 4. The decay heat generation rates for fuel from each previous discharge cycle are summarized below. It was conservatively assumed that the EFPD for all previously discharged fuel assemblies was 1260 days.

Cycle	Unit 1 Decay Heat (BTU/Hr)	Unit 2 Decay Heat (BTU/Hr)
1	182106	239391
2	184518	285822
3	189342	232155
4	197181	232155
5	198990	306928
6	204417	293661
7	261702	297882
8	273159	305118
9	287631	325017
10	300294	335871
11	311751	346725
12	324414	358785
13	337077	370845

14	348534	384714
15	361197	400392
16	373257	422100
17	385920	467928
18	399789	598176
19	414261	1078164
20	434160	
21	472149	
22	566820	
23	773046	
24	2981835	

Table 1 - Discharge Schedule for Unit 1 Reactor						
Cycle	EOC Date	# of Assys	Pool Total	Burnup	Enrich.	Weight U
1A	23-Dec-1976	. 65	65	19,100	2.25	452.6
2A	6-Apr-1978	64	129	29,100	2.80	454.9
3A	6-Apr-1979	64	193	34,200	3.29	451.7
4A	30-May-1980	65	338	31,900	2.93	429.1
5A	29-May-1981	64	494	31,300	2.92	428.4
6A	4-Jul-1982	64	558	31,600	2.90	427.4
7A	17-Jul-1983	80	710	31,400	2.91	427.4
8A	6-Apr-1985	80	882	30,100	2.91	. 427.4
9A	22-Jun-1987	80	1050	35,300	3.19	446.4
10A	19-Mar-1989	80	1210	37,800	3.47	459.8
11A	11-Oct-1990	80	1367	35,600	3.43	460.7
12A	22-Jun-1992	80	1523	39,400	3.37	460.2
13A	12-Feb-1994	80	1603	38,400	3.45	460.8
14A	14-Jul-1995	80	1759	36,700	3.40	461.1
15A	20-Dec-1996	84	1927	52,200	3.50	461.0
16A	25-May-1998	84	2095	52,200	3.50	461.0
17A	28-Oct-1999	84	2263	52,200	3.50	461.0
18A	1-Apr-2001	84	2431	52,200	3.50	461.0
19A	4-Sep-2002	84	2599	52,200	3.50	461.0
20A	7-Feb-2004	84	2767	52,200	3.50	461.0
21A	12-Jul-2005	84	2935	52,200	3.50	461.0
22A	15-Dec-2006	84	3103	52,200	3.50	461.0
23A	19-May-2008	84	3271	52,200	3.50	461.0
24A	22-Oct-2009	84	3439	52,200	3.50	461.0
25A	27-Mar-2011	84	3607	52,200	3.50	461.0

Table 2 - Discharge Schedule for Unit 2 Reactor						
Cycle	EOC Date	# of Assys	Pool Total	Burnup	Enrich.	Weight U
1B	20-Oct-1979	80	273	16,600	2.16	459.3
2B	15-Mar-1981	92	430	28,200	2.84	459.6
3B	22-Nov-1982	72	630	32,100	3.38	459.4
4B	10-Mar-1984	92	802	34,300	3.27	459.2
5B	28-Fab-1986	88	970	36,100	3.57	419.9
6B	1-May-1988	80	1130	39,400	3.64	402.2
7B	30-Jun-1990	77	1287	40,200	3.77	402.7
8B	20-Feb-1992	76	1443	42,500	3.96	402.9
9B	6-Sep-1994	76	1679	43,500	3.82	412.7
10B	15-Mar-1996	84	1843	64,800	4.00	410.0
11B	23-Aug-1997	84	2011	64,800	4.00	410.0
12B	26-Jan-1999	84	2179	64,800	4.00	410.0
13B	30-Jun-2000	84	2347	64,800	4.00	410.0
14B	3-Dec-2001	84	2515	64,800	4.00	410.0
15B	8-May-2003	84	2683	64,800	4.00	410.0
16B	10-Oct-2004	84	2851	64,800	4.00	410.0
17B	15-Mar-2006	84	3019	64,800	4.00	410.0
18B	18-Aug-2007	84	3187	64,800	4.00	410.0
19B	20-Jan-2009	84	3355	64,800	4.00	410.0
20B	25-Jun-2010	84	3523	64,800	4.00	410.0

Table 3 - Maximum SFP Bulk Temperature and Coincident Heat Loads and Losses					
Case Number	Maximum SFP Temperature (°F)	Coincident Time After Reactor Shutdown (hours)	Coincident Heat Load to SPF HXs (MBtu/hr)	Coincident Evaporation Heat Losses (MBtu/hr)	Number of Cooling Trains
Case1A	154.37	138.0	27.19	2.35	1
Case 1B	151.39	137.0	27.55	2.04	1
Case 2	128.68	131.0	29.32	0.57	2
Case 3	146.88	155.0 *	54.27	1.63	2
Case 4	180.75	157.0 *	47.32	8.40	1

* For Case 3 and Case 4, the coincident time is measured from the shutdown of the Unit 2 reactor (second discharge).

Table 4 - Boiling Times and Maximum Evaporation Rates		
Case Number	Time to Start of Boiling (hours)	Maximum Evaporation Rate (gal/min)
Case1A	9.45	63.35
Case1B	9.89	63.40
Case2	13.37	63.64
Case3	6.08	103.68
Case4	2.95	103.85

Table 5 - Maximum Local Pool Water and Fuel Cladding Temperature (Case1A)		
	Max Local Pool Water Temp (°F)	Max Local Fuel Cladding Temp (°F)
No Blockage	163.6	214.5
50% Blockage	223.5	254.3

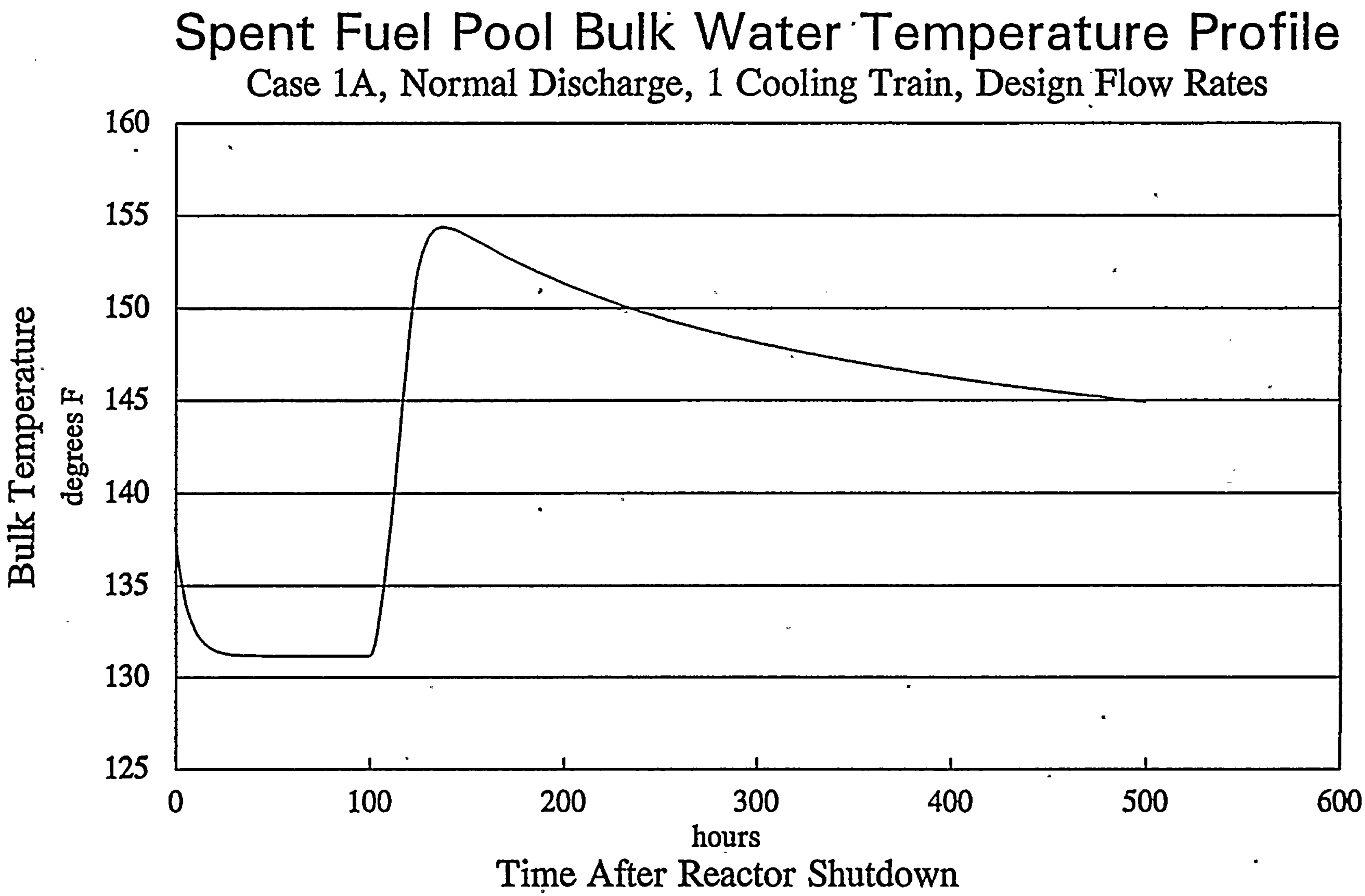

Figure 1: SFP Bulk Water Temperature Profile for Case 1A

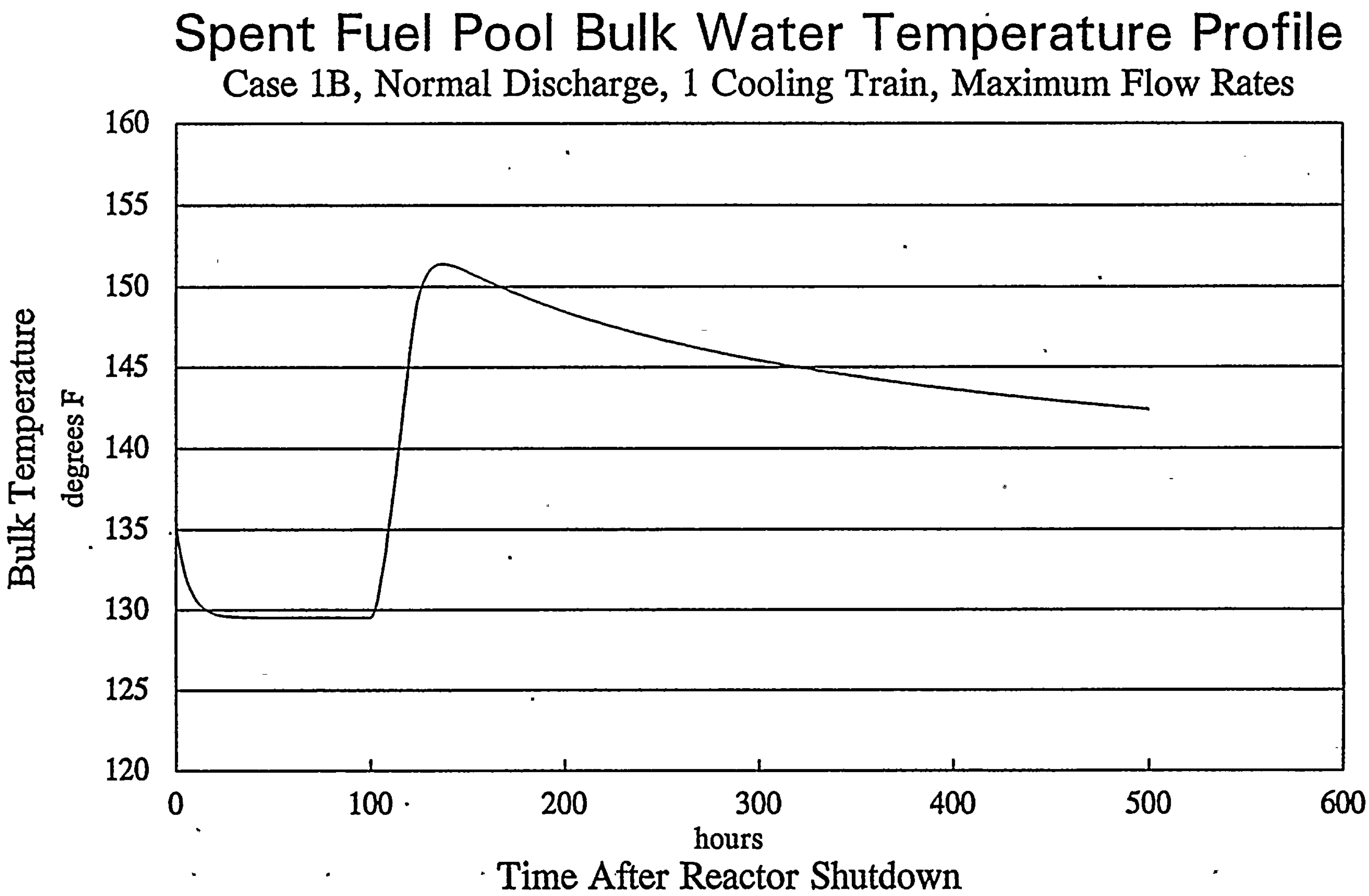

Figure 2: SFP Bulk Water Temperature Profile for Case 1B

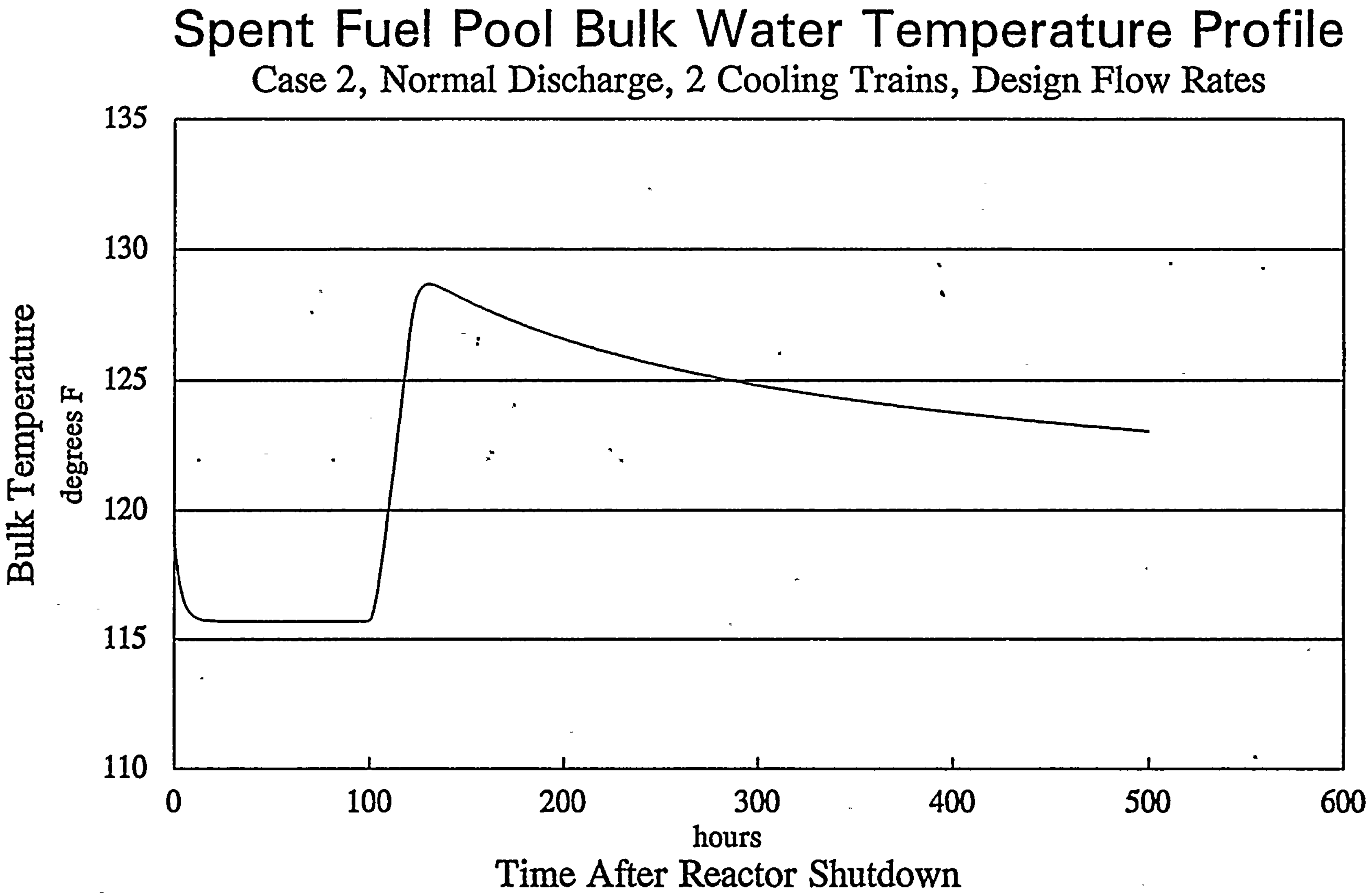

Figure 3: SFP Bulk Water Temperature Profile for Case 2

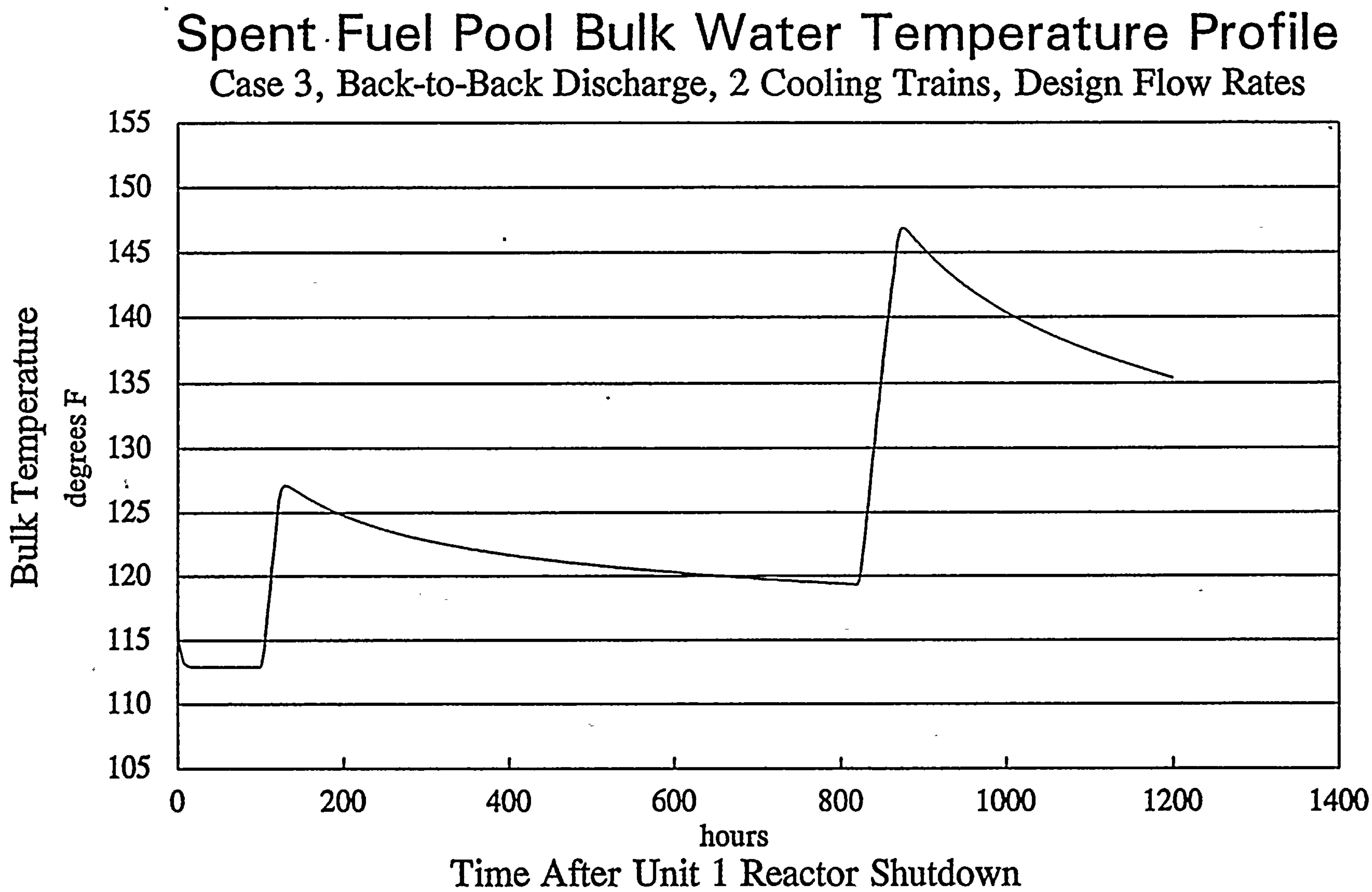

Figure 4: SFP Bulk Water Temperature Profile for Case 3

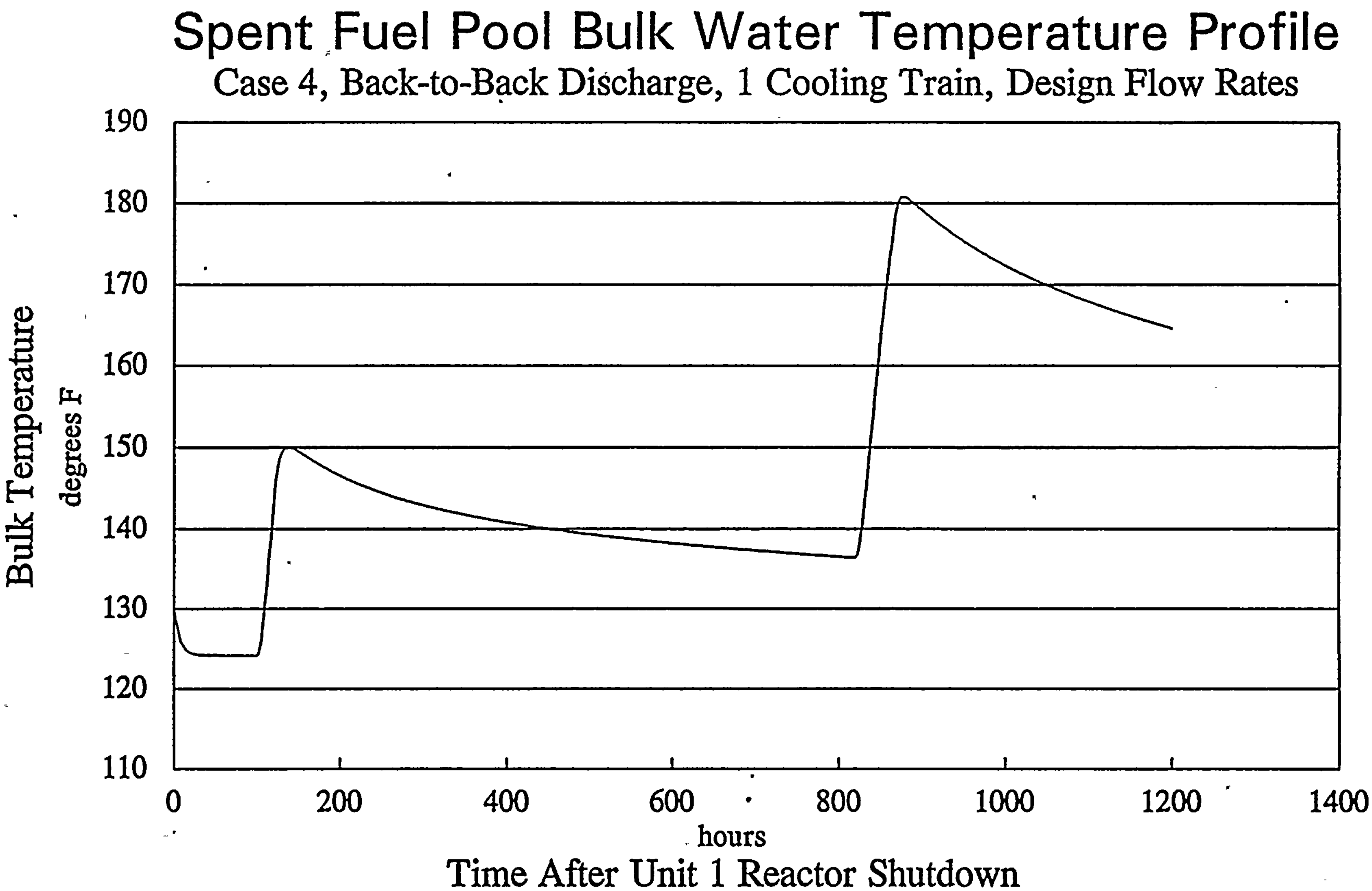

Figure 5: SFP Bulk Water Temperature Profile for Case 4

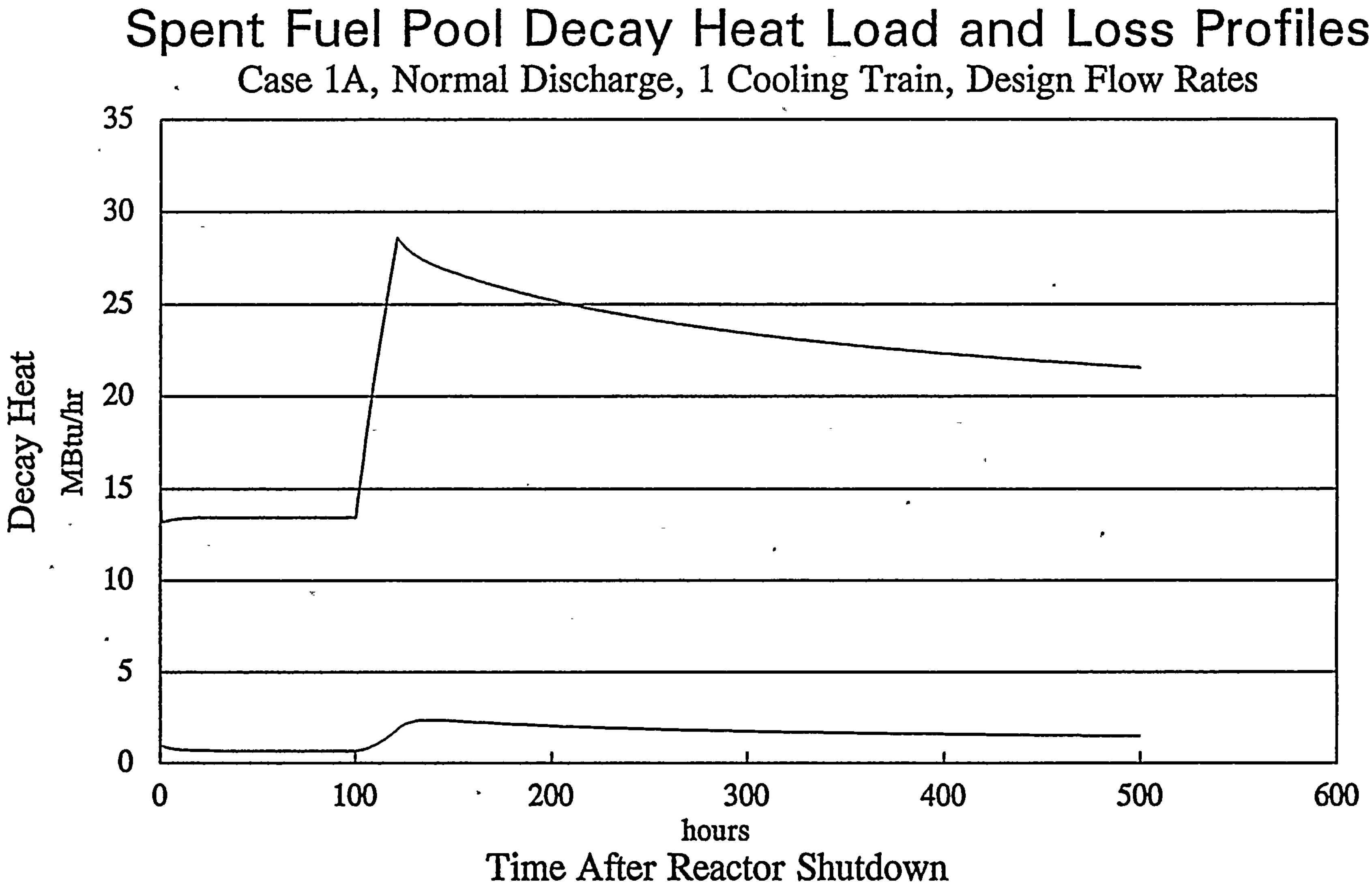

Figure 6: SFP Decay Heat Load and Loss Profiles for Case 1A

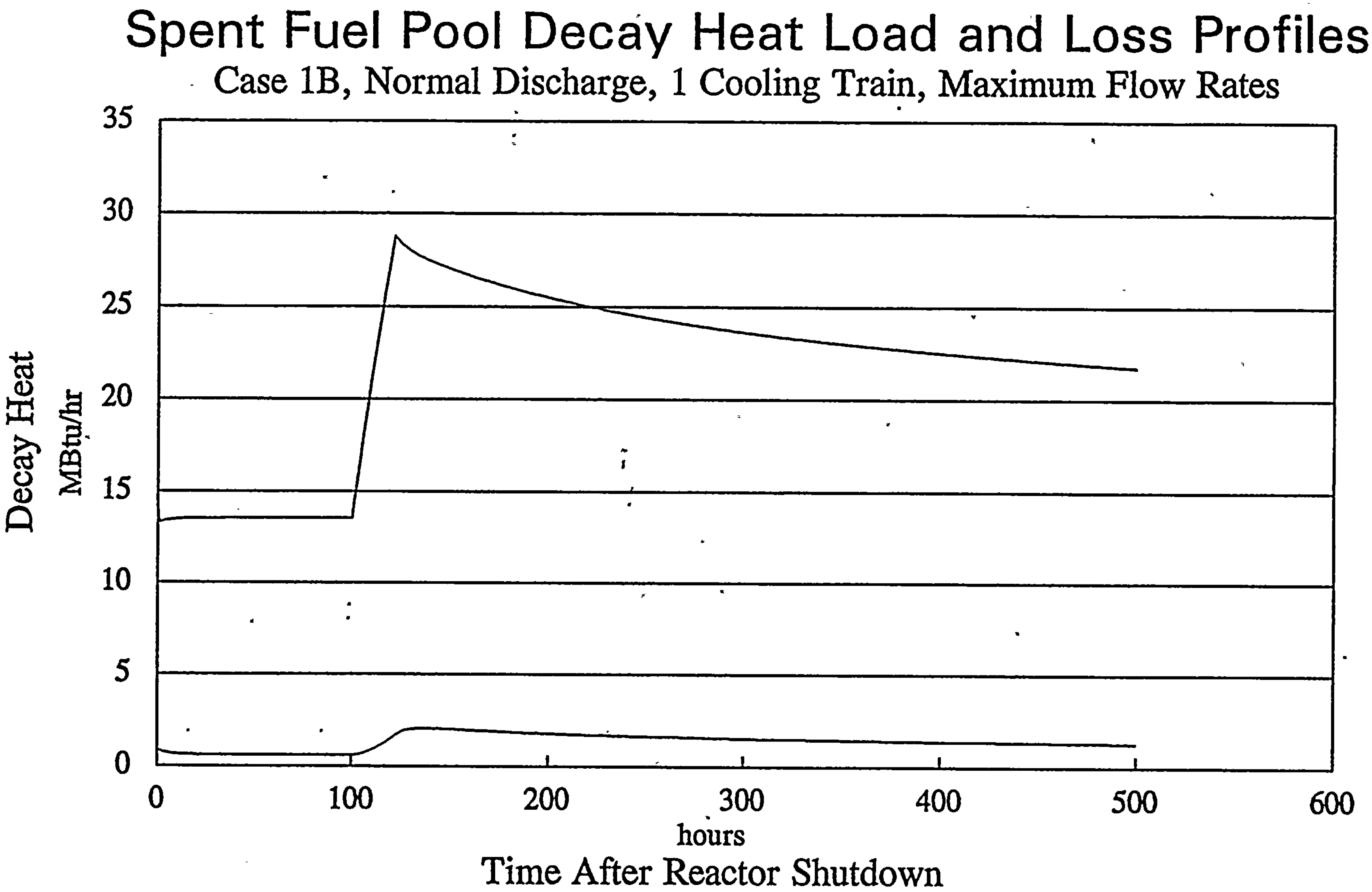

Figure 7: SFP Decay Heat Load and Loss Profiles for Case 1B

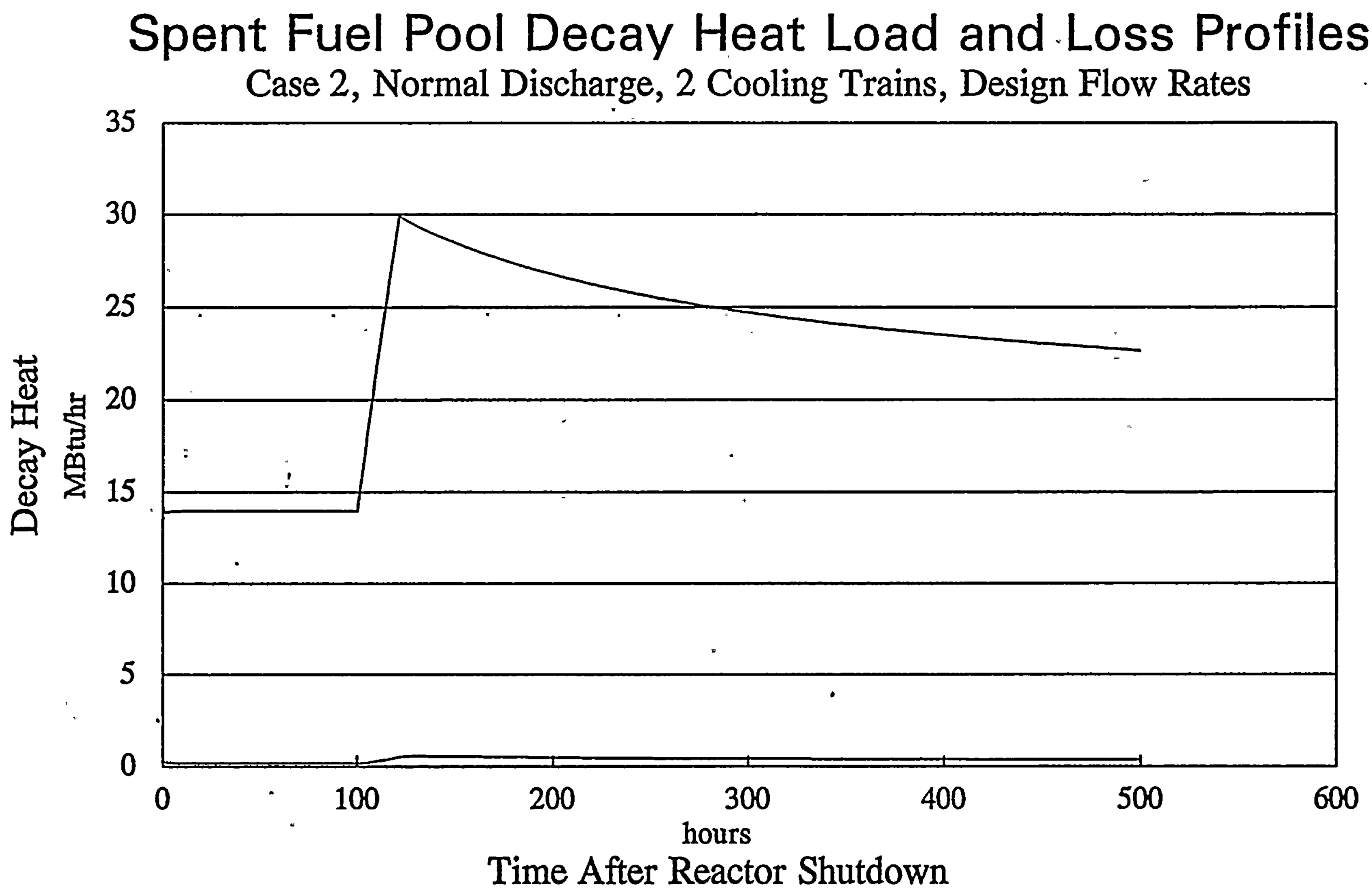

Figure 8: SFP Decay Heat Load and Loss Profiles for Case 2

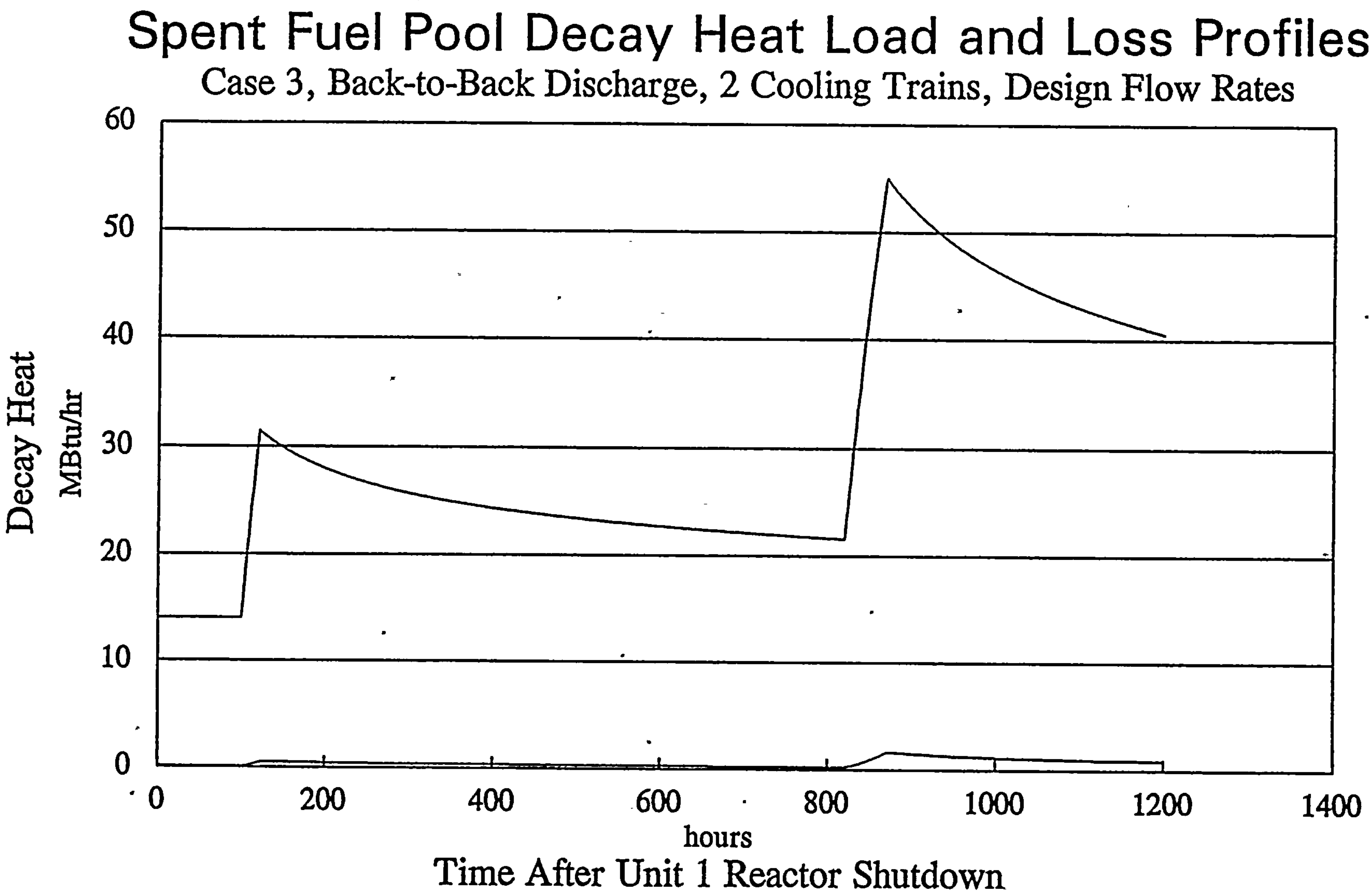

Figure 9: SFP Decay Heat Load and Loss Profiles for Case 3

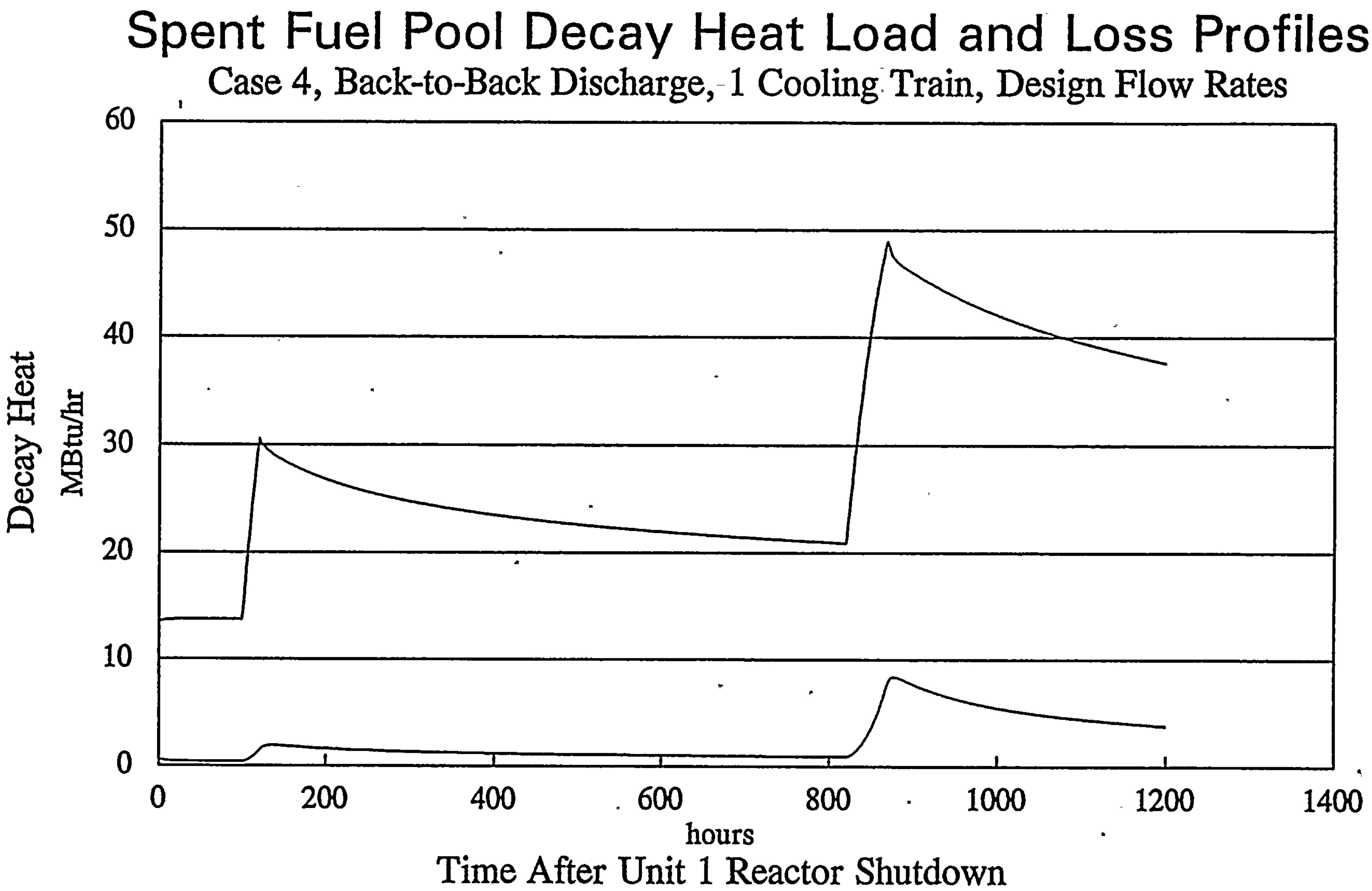

Figure 10: SFP Decay Heat Load and Loss Profiles for Case 4