HOLTEC
INTERNATIONAL

SPENT NUCLEAR FUEL POOL THERMAL-HYDRAULIC ANALYSIS REPORT for DONALD C. COOK NUCLEAR PLANT INDIANA MICHIGAN POWER COMPANY

by

HOLTEC INTERNATIONAL

HOLTEC PROJECT 40224
HOLTEC REPORT HI-941183
REPORT CATEGORY: I

AUGUST, 1994

SUMMARY OF REVISIONS LOG HOLTEC REPORT HI-941183	
REVISION 0 CONTAINS THE FOLLOWING PAGES:	
Title Page	1
Review and Certification Log	1
Summary of Revisions Log	1
Section 1	1
Section 2	24
Section 3	3
Section 4	1
REVISION 1 CONTAINS THE FOLLOWING PAGES:	
Title Page	1
Review and Certification Log	1
Summary of Revisions Log	1
Section 1	1
Section 2	25
Section 3	4
Section 4	1

SUMMARY OF REVISIONS LOG HOLTEC REPORT HI-941183	
REVISION 2: Revisions made to pages 3-1, 3-2, and 3.3. Revision 2 contains the following number of pages:	
Title Page	1
Review and Certification Log	1
Summary of Revisions Log	1
Section 1	1
Section 2	24
Section 3	3
Section 4	1

HOLTEC
INTERNATIONAL

REVIEW AND CERTIFICATION LOG

DOCUMENT NAME:	SPENT NUCLEAR FUEL POOL THERMAL-HYDRAULIC ANALYSIS REPORT for DONALD C. COOK NUCLEAR PLANT
HOLTEC DOCUMENT I.D. NUMBER:	HI-941183
HOLTEC PROJECT NUMBER:	40224
CUSTOMER/CLIENT:	INDIANA MICHIGAN POWER COMPANY

REVISION BLOCK

ISSUE NUMBER	AUTHOR & DATE	REVIEWER & DATE	QA MANAGER & DATE	APPROVED* & DATE
ORIGINAL	WANG 7/27/94	Alan Soler A.S 7/27/94	for M. Soler 7/28/94	SINGH 7/27/94
REVISION 1	WANG 8/4/94	SINGH 8/4/94	MS 8/4/94	SINGH 8/4/94
REVISION 2	WANG 8/25/94	Alan Soler A.S. 8/25/94	for M. Soler 8/25/94	WANG 8/25/94
REVISION 3				
REVISION 4				
REVISION 5				
REVISION 6				

This document conforms to the requirements of the design specification and the applicable sections of the governing codes.

Note: Signatures and printed names are required in the review block.

* Must be Project Manager or his designee.

1.0 INTRODUCTION

In 1992, Donald C. Cook Nuclear Plant received an operating license amendment allowing the twin reactor pool to be reracked with "poisoned" high density racks to store fuel in a Mixed Zone Three Region arrangement. Under a turnkey contract with Holtec International, Cook Nuclear Plant's owner, Indiana Michigan Power Company, reracked the Cook Nuclear Plant spent fuel pool with 23 free-standing modules containing a total of 3613 storage cells. The object of this submittal is to clarify certain ambiguities in the original Licensing Report submitted in support of the 1992 license amendment request (Amendments 169 for Unit 1 and 152 for Unit 2) and to provide additional flexibility in the plant's ability to discharge fuel into the pool subsequent to a planned (or unplanned) shutdown of a reactor unit.

At the present time, Technical Specification 3/4.9.3 stipulates a minimum incore decay after core subcriticality of 168 hours before any transfer of fuel assemblies from the reactor to the spent fuel pool. Considerations of efficient outage management warrant that the plant staff initiate, at its option, fuel transfer 100 hours after core subcriticality. This submittal provides a summary of the analyses carried out to demonstrate the acceptability of reduction of incore decay time from 168 hours to 100 hours.

Reducing the incore decay time prior to discharging the spent fuel to the spent fuel pool entails a potential change in the pool bulk temperature. Inasmuch as the pool bulk temperature affects the thermal moment and shear in the reinforced concrete structure, it is necessary to determine the impact of the proposed change on the pool structure as well. Computations to establish continued compliance of the pool structure to the applicable regulatory requirements are also summarized herein.

The minor changes to the Licensing Report pertain to clarifying the Boral in-service inspection program, and editorial changes to the number of cells ascribed to Regions 1, 2 and 3 of the Licensing Report [1] are also included in this report.

2.0 THERMAL-HYDRAULIC EVALUATION

The thermal-hydraulic considerations documented in Section 5.0 of Ref. [1] are repeated in this submittal to reflect the changes in (1) the minimum incore decay time and (2) minor revision of the refueling discharge schedule for both units at Cook Nuclear Plant. The methodology and computer codes used in this submittal are identical to those of Ref. [1]. The analysis procedures are summarized in Section 2.1; the discharge scenarios are shown in Section 2.2, and the results are presented in Section 2.3.

2.1 Analysis Procedures

The thermal-hydraulic evaluation for the spent fuel pool and the rack array consist of the the following discrete steps:

(i) Evaluation of long term decay heat load, which is the accumulating spent fuel decay heat generation based on the existing and the predicted operating cycles at the time instant of the final refueling cycle according to the storage capacity of the fuel pool. The heat load is treated as constant to combine with the transient decay heat generated by the final discharge.

(ii) Evaluation of the total transient decay heat load including the long term decay heat determined in (i) and the pool bulk temperature as a function of time during the final postulated discharge scenarios.

(iii) Evaluation of the time-to-boil if all forced heat rejection paths from the pool are lost.

(iv) Determination of the maximum pool local temperature at the instant when the bulk temperature reaches its maximum value.

(v) Evaluation of the maximum fuel cladding temperature to establish that bulk nucleate boiling at any location around the fuel is not possible with cooling available.

(vi) Compute the effect of a blocked fuel cell opening on the local water and maximum cladding temperature.

2.2 Discharge Scenario

The revised existing and projected spent fuel discharge schedules for D. C. Cook spent fuel pool from both units are shown in Table 2.1. The decay heat generation rate in the pool is computed using this data. All discharge scenarios considered herein are intended to be predicated on the maximum residual heat load from previously discharged fuel. Accordingly, all four discharge scenarios (Case 1 through 4 below) are considered during a refueling outage close to the end of the licensed storage capacity of 3613 cells, when the pool has the highest decay heat generation rate from the old fuel stored in the pool. Since the decay heat generation generally depends on both the total number of assemblies in the pool and the decay time of the last discharged batch, three candidate instances of maximum decay heat load exist. Calculations are performed for the decay heat during the refueling of cycle 20B (Unit 2 cycle 20), 25A (Unit 1 Cycle 25), and 21b because they feature different combinations of the total number assemblies and the time duration between the outages. The results indicate that the pool has slightly higher decay heat generation rate from the previously discharged fuel during cycle 20B refueling in December, 2009, compared to the two other candidate cases, and therefore, the discharge scenarios will be considered during this outage. Please note that this analysis bounds the conditions up to Cycle 21b, when a hypothetical maximum 3824 spent fuel assemblies will be in the pool after a back-to-back full core offload. In this manner, this analysis provides conservative thermal-hydraulic calculation for the entire storage life.

The size of the normal discharge batch is assumed to be 80 assemblies, as was the case in the rerack licensing submittal.

CASE 1 - Normal Discharge, Single Train

In cycle 20B refueling (from Unit 2), a total of 80 assemblies are discharged to the pool. The fuel transfer starts 100 hours after reactor shutdown and transfers to the pool at the rate of 4 assemblies per hour. All the fuel discharged are assumed to have 1260 EFPD of operation at a rated power of 3411 MW in the reactor. One of the two spent fuel pool

cooling trains is running to cool the pool. The case is also analyzed for actual measured SFP flow of 2800 gpm. The results corresponding to design basis flow (2300 gpm) and 2800 gpm (actual measures) are labeled as Case 1A and 1B, respectively. The design basis flow rates are used for all other cases. A maximum of 3399 assemblies (assume 80 instead of 76 assemblies discharged in this batch 20B) are considered in this case.

CASE 2 - Normal Discharge, Both Trains

Same as Case 1 except for that two cooling trains are available. Figure 2.1 schematically shows the normal discharge.

CASE 3 - Back-To-Back Full Core Offload, Both Trains

The Unit 1 reactor has an unplanned shutdown 30 days after the Unit 2 shutdown. A full core of 193 assemblies are discharged to the pool after the Unit 2 normal discharge. The full core offload starts 100 hours after reactor shutdown and transfers fuel assemblies to the pool at the rate of 4 assemblies per hour. The average burnup of the core is assumed to be that 80 assemblies have 420 EFPD of operation in the reactor, and the remaining 113 assemblies are assumed to have 1260 EFPD of operation. Two spent fuel pool cooling trains are running to cool the pool. Figure 2.2 schematically shows the discharge. A maximum of 3592 assemblies are considered in this scenario.

CASE 4 - Back-To-Back Full Core Offload, Single Train

Same as Case 3 except only one cooling train is in operation. This case is *not* a design basis scenario for Cook Nuclear Plant or the USNRC guidelines (NUREG-0800). It is presented for reference purposes only.

2.3 Analysis Results

The calculated maximum accumulating long term decay heat during the outages close to the end of the fuel pool storage capacity is 18.15 x 10^6 Btu/hr based on the discharge projections shown in Table 2.1. The maximum number of cycles considered is based on the maximum storage capacity of 3613 cells. The maximum bulk pool temperature results and the heat loads at the instant of maximum temperature are presented in Table 2.2. The time varying bulk pool temperatures and heat loads in the pool are plotted vs. time-after-shutdown in Figures 2.3 to 2.12. It is shown from the analyses that the spent fuel pool cooling system has sufficient cooling capacity to maintain the spent fuel pool bulk water temperature at or below 161°F (Case 1A) during a normal refueling discharge (80 assemblies), with one or two cooling trains operating, and the net normal heat load, coincident to the maximum water temperature, is 30.8 x 10^6 Btu/hr(excluding evaporation heat losses). Two trains of the spent fuel pool cooling system have sufficient heat removal capacity to maintain the spent fuel pool bulk water temperature below 151°F (Case 3) during an assumed back-to-back full core offload and the coincident abnormal heat load is 58.7 x 10^6 Btu/hr (excluding evaporation heat losses).

As shown in Table 2.2, the previous licensing basis analysis indicated that the maximum normal water temperature was 160°F. The previous net normal heat load coincident to the maximum water temperature was 30.2 x 10^6 Btu/hr(excluding evaporation heat losses). Comparison with the previous rerack submittal analysis bulk pool temperature results (also provided in Table 2.2) shows that the proposed thermal-hydraulic changes have insignificant thermal consequences. The previous maximum abnormal water temperature was 144°F during an assumed back-to-back full core offload. The previous coincident abnormal heat load was 50.7 x 10^6 Btu/hr (excluding evaporation heat losses).

The loss-of-cooling events have also been considered for the specified discharge scenarios. The loss of all forced cooling is conservatively assumed to occur at the instant of peak pool temperature. Table 2.3 summarizes the results of the time-to-boil and maximum evaporation rate under the conservative assumption that no makeup water is provided to the pool. The

calculated minimum time from the loss-of-pool cooling until the pool boils for the design bases case is 4.51 hours (Case 3) and the maximum boiloff rate is 129.23 gpm during the full core offload. The time-to-boil is 7.28 hrs and maximum boiloff rate is 72.22 gpm during the design basis normal discharge.

Consistent with our approach to make the most conservative assessments of temperature, the local water temperature calculations are performed assuming that the pool is at its peak bulk temperature. Thus, the local water temperature evaluation is, in essence, calculation of the temperature increment over the theoretical spatially uniform value due to local hot spots (due to the presence of a highly heat emissive fuel bundle).

The maximum local water temperature for the limiting case (Case 1A) is calculated to be 171.9°F and the maximum local fuel cladding temperature is 224.4°F. If the limiting cells are 50% blocked on the top, the maximum local water temperature becomes 231.5°F and the maximum fuel cladding temperature is 264.2°F (see Table 2.4). The local boiling point at the depth of 23 ft of water is 238°F. Therefore, nucleate boiling will not occur even around the fuel rods, even under conditions of maximum postulated heat flux.

2.4 Effect on Pool Structure

It is recalled from the rerack licensing submittal that the structural evaluation of the spent fuel pool reinforced concrete structure was based on a temperature differential, ΔT, of 85°F between the inside and outside faces of the pool structure. A thermal heat flow path analysis across the reinforced concrete sections for the highest peak pool bulk temperature case shows ΔT to be 69°F. Therefore, the margins of safety for the pool structure reported in the rerack submittal continue to bound the actual conditions.

2.5 Conclusion

The foregoing results indicate that the maximum bulk spent fuel pool water temperature is increased by 1°F from the previous 160°F to 161°F. Therefore, the margin of safety established in the original rerack license submittal [1] has not been significantly reduced.

Table 2.1 FUEL CYCLE AND SPENT FUEL DISCHARGE SUMMARY UNIT 1						
Cycle	BOC Date	EOC Date	Cycle EFPD	Discharge Assemblies	Cumulative Discharge Into Pool from Unit 1	Total Pool Inventory
1A	18-Jan-75	23-Dec-76	463	65	65	65
2A	20-Feb-77	06-Apr-78	284	64	129	129
3A	18-Jun-78	06-Apr-79	257	64	193	193
4A	08-Jul-79	30-May-80	268	65	258	338
5A	04-Aug-80	29-May-81	217	64	322	494
6A	01-Aug-81	04-Jul-82	264	64	386	558
7A	16-Sept-82	17-Jul-83	265	80	466	710
8A	21-Oct-83	06-Apr-85	410	80	546	882
9A	17-Nov-85	22-Jun-87	425	80	626	1050
10A	05-Oct-87	19-Mar-89	428.5	80	706	1210
11A	30-Jun-89	11-Oct-90	437	80	786	1367
12A	23-Jan-91	22-Jun-92	459	80	866	1523

Table 2.1 (continued) FUEL CYCLE AND SPENT FUEL DISCHARGE SUMMARY UNIT 1						
Cycle	BOC Date	EOC Date	Cycle EFPD	Discharge Assemblies	Cumulative Discharge Into Pool from Unit 1	Total Pool Inventory
13A	28-Oct-92	12-Feb-94	445	80	946	1603
14A	11-May-94	05-Jul-95	420	80	1026	1759
15A	02-Nov-95	26-Dec-96	420	80	1106	1915
16A	21-Mar-97	15-May-98	420	80	1186	2071
17A	08-Aug-98	02-Oct-99	420	80	1266	2227
18A	26-Dec-99	18-Feb-01	420	80	1346	2383
19A	14-May-01	08-Jul-02	420	80	1426	2539
20A	01-Oct-02	25-Nov-03	420	80	1506	2695
21A	24-Mar-04	18-May-05	420	80	1586	2851
22A	11-Aug-05	05-Oct-06	420	80	1666	3007
23A	29-Dec-06	22-Feb-08	420	80	1746	3163
24A	17-May-08	11-Jul-09	420	80	1826	3319
25A	04-Oct-09	28-Nov-10	420	80	1906	3475

Table 2.1 (continued)

FUEL CYCLE AND SPENT FUEL DISCHARGE SUMMARY

UNIT 2

Cycle	BOC Date	EOC Date	Cycle EFPD	Discharge Assemblies	Cumulative Discharge Into Pool from Unit 2	Total Pool Inventory
1B	10-Mar-78	20-Oct-79	396	80	80	273
2B	18-Jan-80	15-Mar-81	335	92	172	430
3B	19-May-81	22-Nov-82	453	72	244	630
4B	21-Jan-83	10-Mar-84	337	92	336	802
5B	07-Jul-84	28-Feb-86	406	88	424	970
6B	11-Jul-86	01-May-88	428	80	504	1130
7B	17-Mar-89	30-Jun-90	407	77	581	1287
8B	10-Nov-90	20-Feb-92	420	76	657	1443
9B	17-Dec-92	02-Sep-94	428	76	733	1679
10B	26-Nov-94	20-Jan-96	420	76	809	1835
11B	14-Apr-96	08-Jun-97	420	76	885	1991
12B	01-Sep-97	26-Oct-98	420	76	961	2147
13B	19-Jan-99	14-Mar-00	420	76	1037	2303
14B	12-Jul-00	05-Sep-01	420	76	1113	2459
15B	29-Nov-01	23-Jan-03	420	76	1189	2615
16B	18-Apr-03	11-Jun-04	420	76	1265	2771

Table 2.1 (continued)

FUEL CYCLE AND SPENT FUEL DISCHARGE SUMMARY

UNIT 2

Cycle	BOC Date	EOC Date	Cycle EFPD	Discharge Assemblies	Cumulative Discharge Into Pool from Unit 2	Total Pool Inventory
17B	04-Sep-04	29-Oct-05	420	76	1341	2927
18B	22-Jan-06	18-Mar-07	420	76	1417	3083
19B	11-Jun-07	04-Aug-08	420	76	1493	3239
20B	28-Oct-08	22-Dec-09	420	76	1569	3395
21B	21-Apr-10	15-Jun-11	420	76	1645	3551

Table 2.2 MAXIMUM SFP BULK POOL TEMPERATURE AND COINCIDENT TIME						
Case Number and Description	Maximum Pool Temp., °F		Present Coincident Time After Reactor Shutdown, hrs.	Present Coincident Heat Load to SFP HXs 10^6 Btu/hr	Present Coincident Evaporation Heat Losses 10^6 Btu/hr	Number of Cooling Trains
	Present Submittal	Previous Value				
1A (normal discharge, Design Basis Flow)	160.48	159.54	136	30.84	3.14	1
1B (normal discharge, actual S.F. water flow)	157.25	156.31	136	31.28	2.70	1
2 (normal discharge, Design Basis flow)	132.26	131.57	129	33.62	0.72	2
3 (Back-to-back full core offload)	150.57	143.84	155	58.66	1.96	2
4 (same as 3, reference case only)	185.07	176.91	156	49.87	10.65	1

Table 2.3 RESULTS OF LOSS-OF-COOLING (No Makeup Water Assumed)			
Case Number	Time Required for Operator action (hours)		New Maximum Evaporation Rate (GPM)
	New Computed Value	Existing Submittal	
1A	7.28	7.82	72.22
1B	7.72	8.27	72.27
2	10.58	11.52	72.56
3	4.51	5.74	129.23
4	1.98	3.02	129.55

Table 2.4 MAXIMUM LOCAL POOL WATER AND FUEL CLADDING TEMPERATURE FOR THE LIMITING CASE (CASE 1A)		
	Maximum Local Pool Water Temp., °F	Maximum Local Fuel Cladding Temp., °F
No Blockage	171.9	224.4
50% Blockage	231.5	264.2

HOLTEC INTERNATIONAL

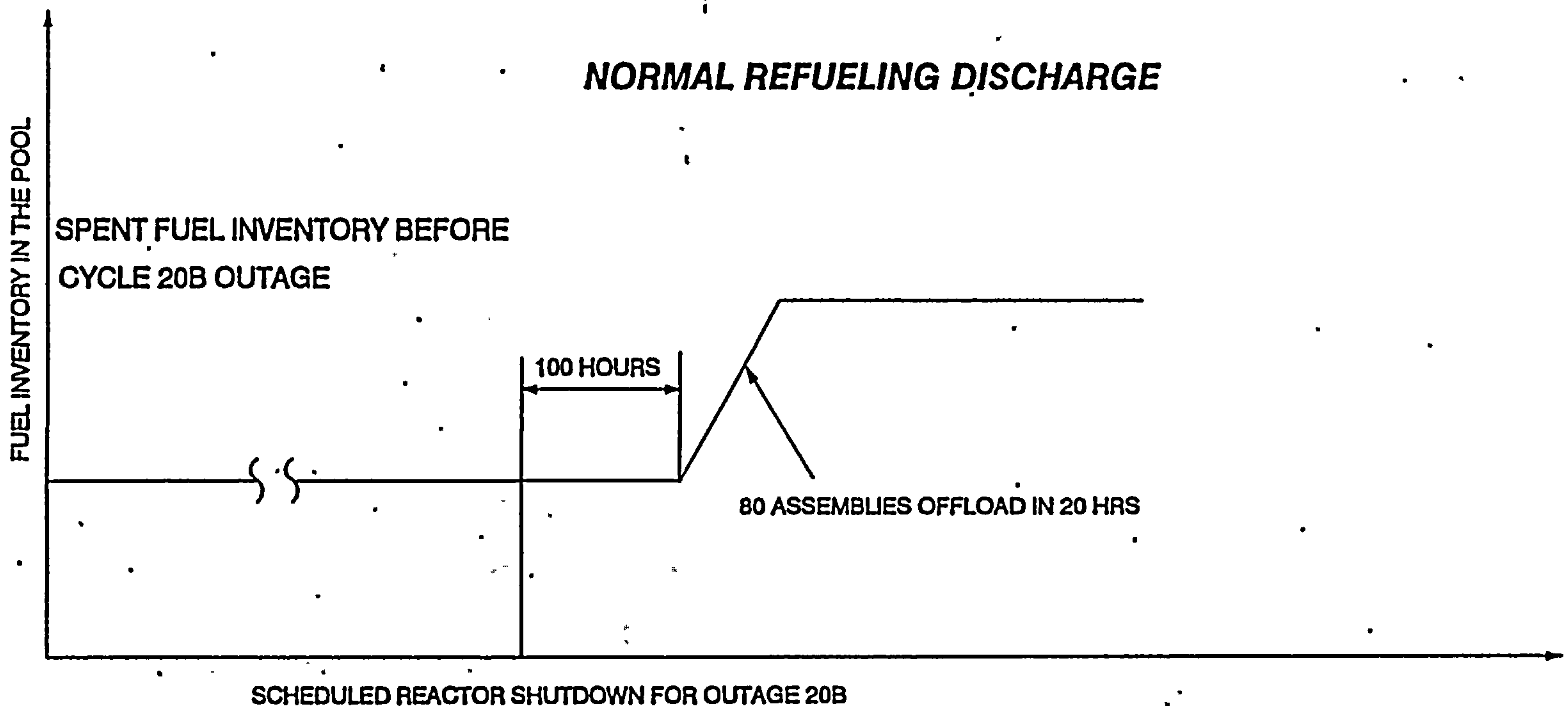

FIGURE 2.1 DONALD C. COOK SPENT FUEL POOL DISCHARGE SCENARIO CASES 1 & 2

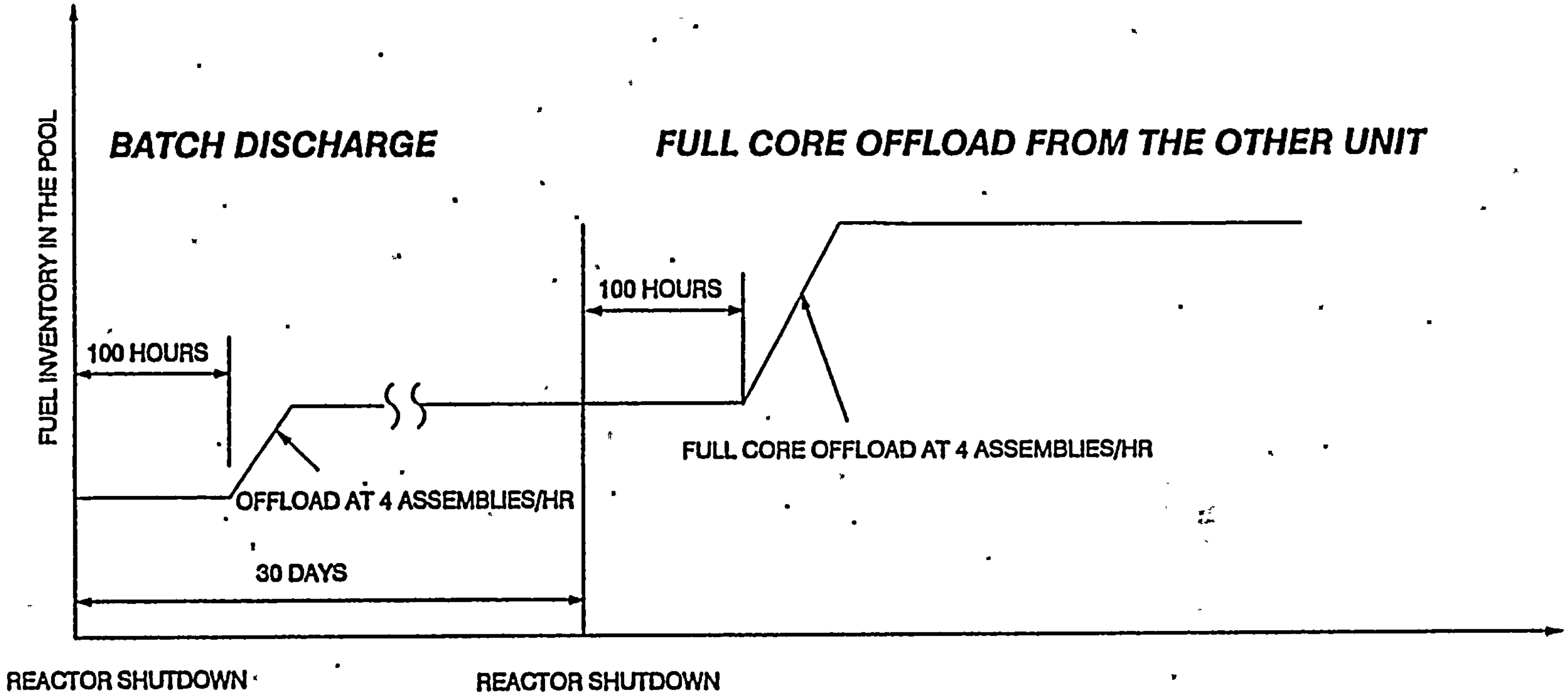

FIGURE 2.2 DONALD C. COOK SPENT FUEL POOL DISCHARGE SCENARIO CASES 3 & 4

HOLTEC INTERNATIONAL

DONALD C. COOK SPENT FUEL POOL
NORMAL DISCHARGE (80 ASSEMBLIES) 2300 GPM SFP FLOW
ONE COOLING TRAIN, CASE 1A

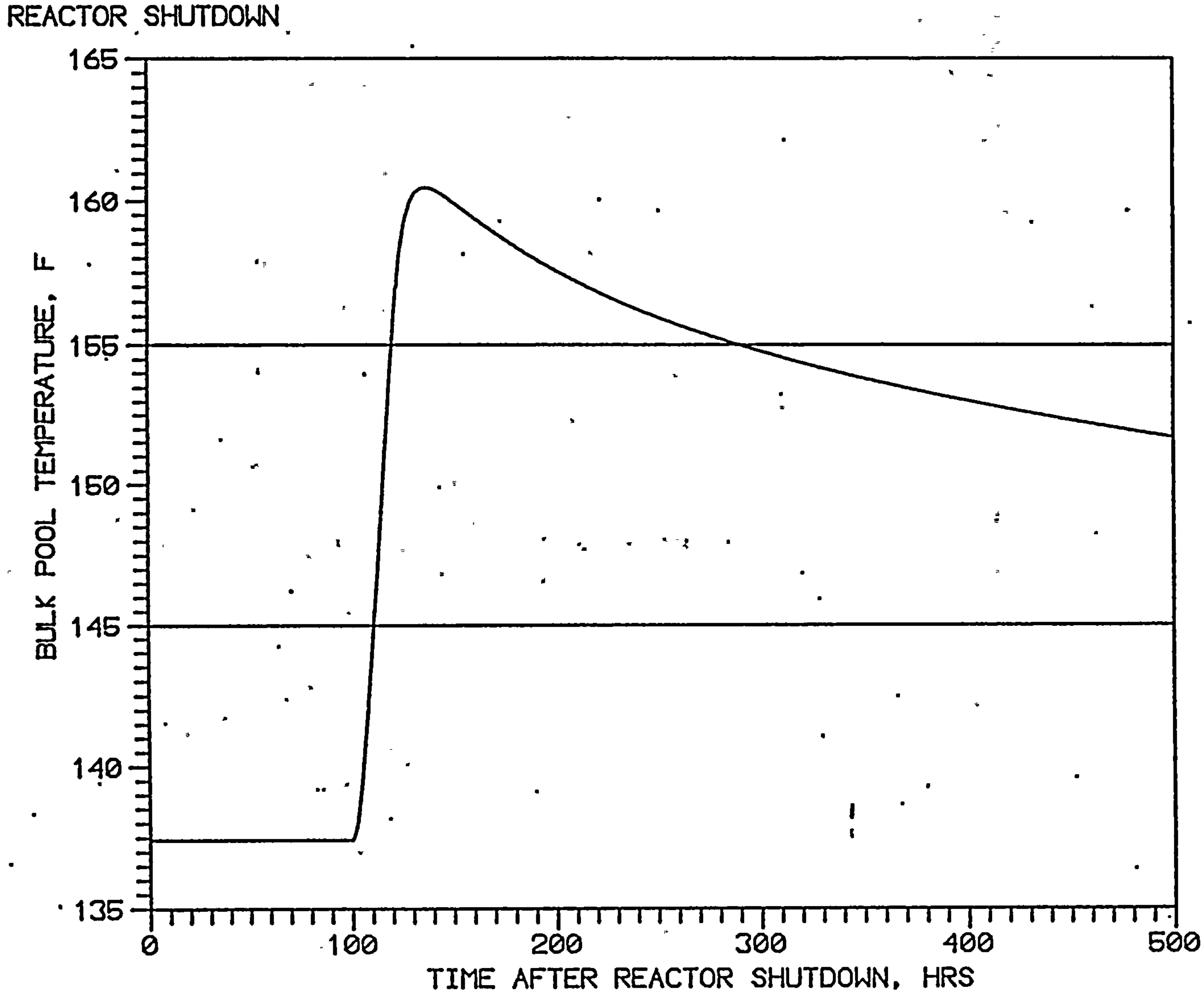

FIGURE 2.3 SFP BULK WATER TEMPERATURE PROFILE

HOLTEC INTERNATIONAL

DONALD C. COOK SPENT FUEL POOL
NORMAL DISCHARGE (80 ASSEMBLIES) 2800 GPM SFP FLOW
ONE COOLING TRAIN, CASE 1B

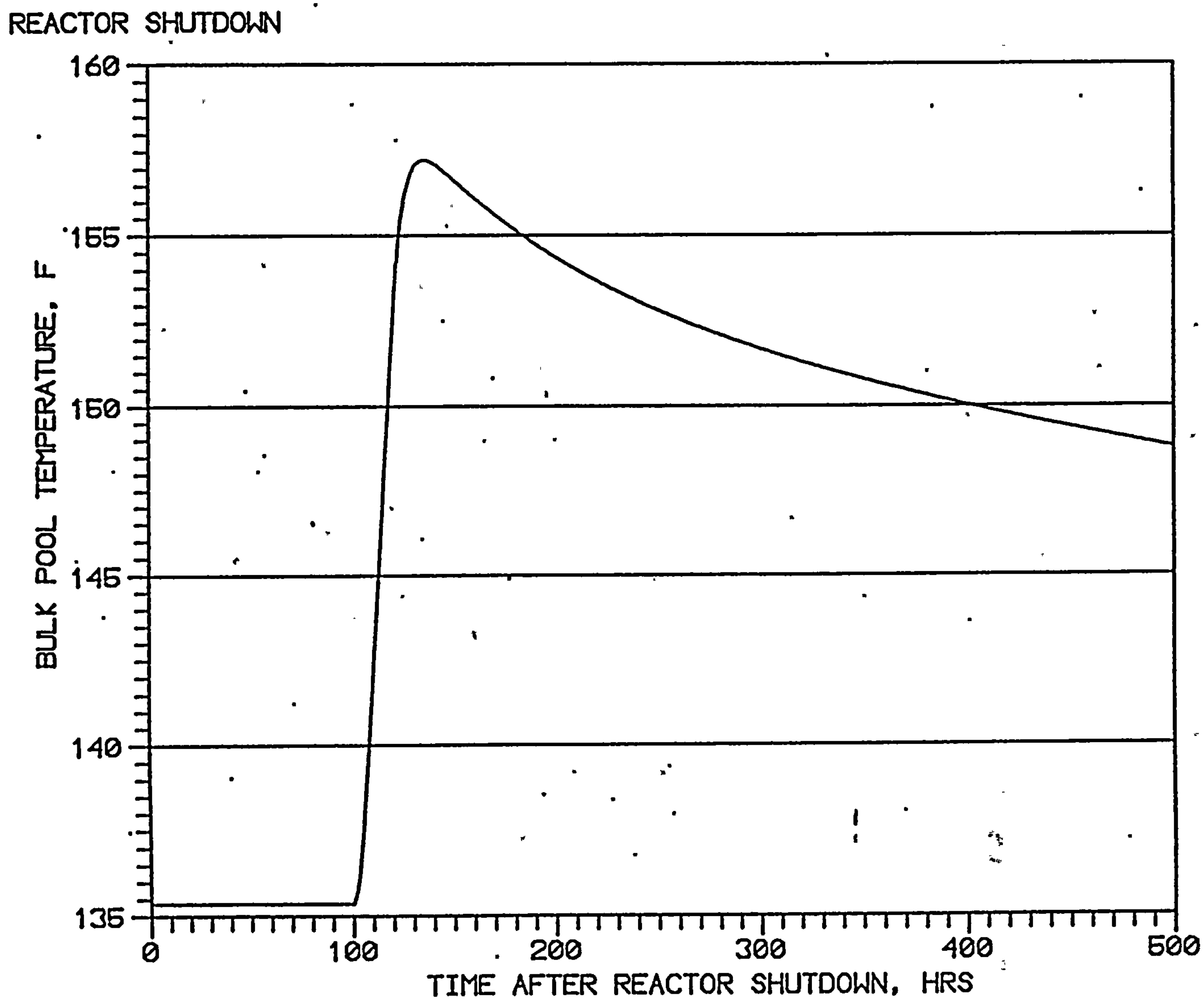

FIGURE 2.4 SFP BULK WATER TEMPERATURE PROFILE

HOLTEC INTERNATIONAL

DONALD C. COOK SPENT FUEL POOL
NORMAL DISCHARGE (80 ASSEMBLIES) 2300 GPM SFP FLOW
TWO COOLING TRAINS, CASE 2

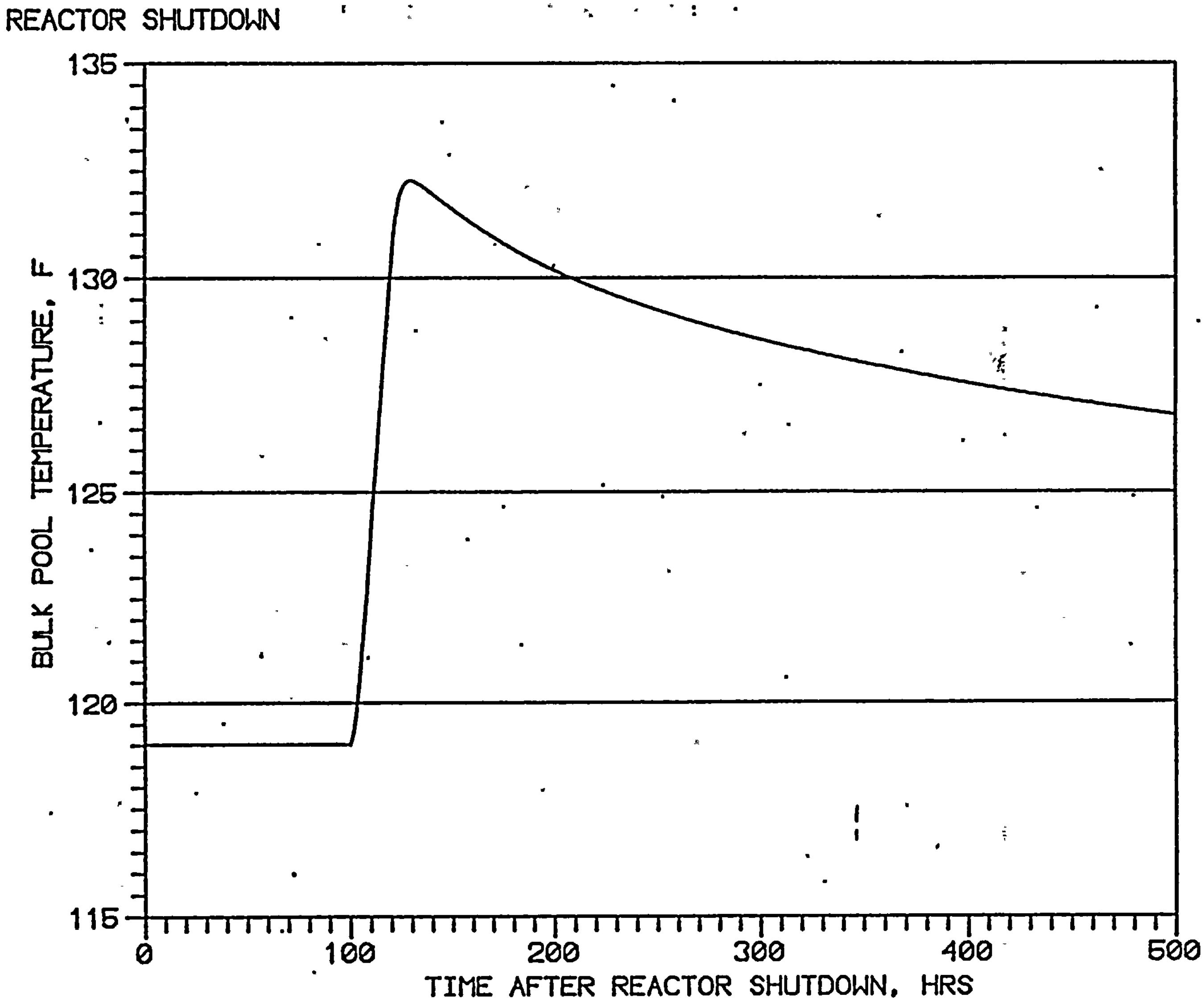

FIGURE 2.5 SFP BULK WATER TEMPERATURE PROFILE

HOLTEC INTERNATIONAL

DONALD C. COOK SPENT FUEL POOL
BACK-TO-BACK FULL CORE OFFLOAD, 2300 GPM SFP FLOW/COOLER
TWO COOLING TRAINS, CASE 3

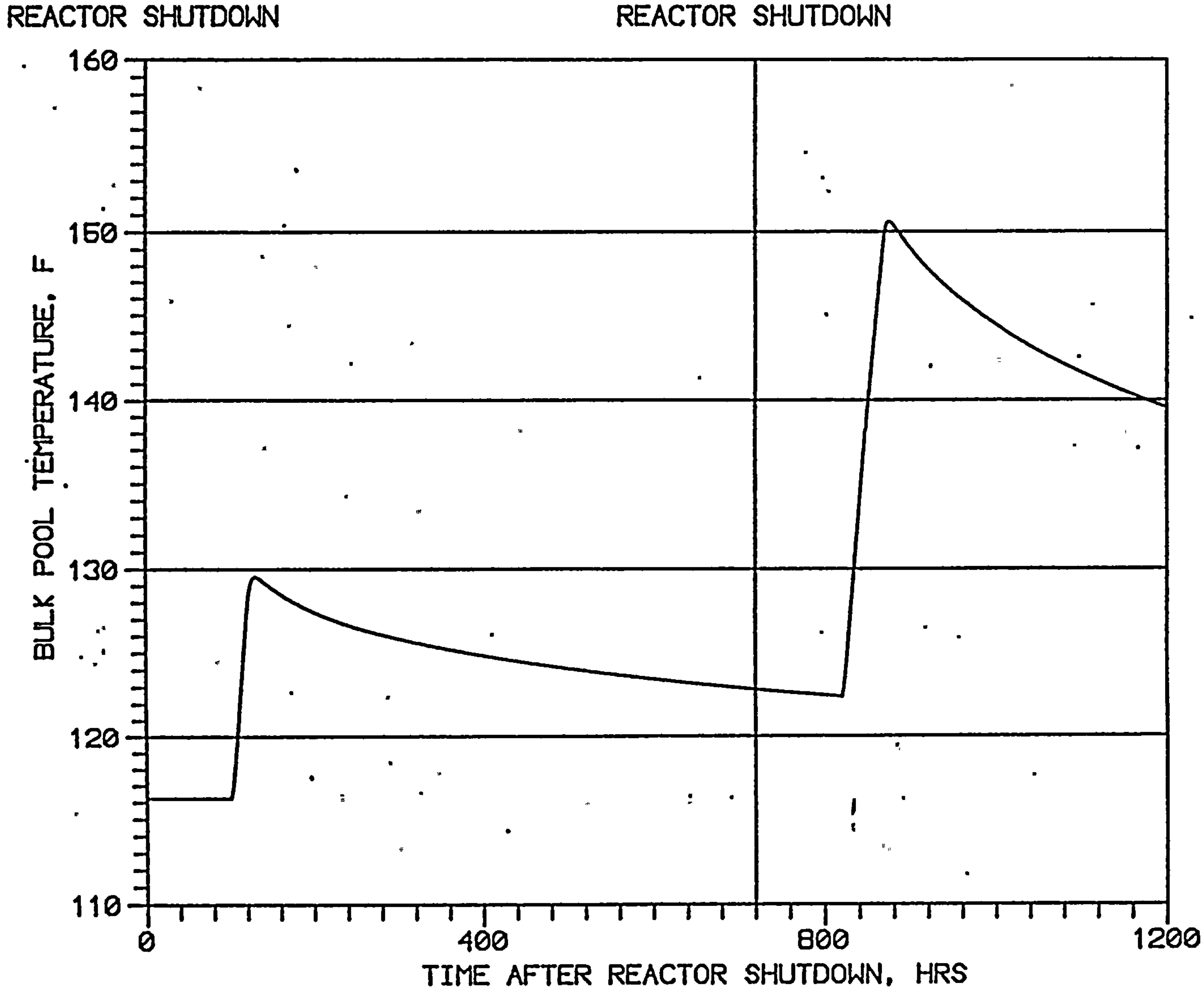

FIGURE 2.6 SFP BULK WATER TEMPERATURE PROFILE

HOLTEC INTERNATIONAL

DONALD C. COOK SPENT FUEL POOL
BACK-TO-BACK FULL CORE OFFLOAD, 2300 GPM SFP FLOW/COOLER
ONE COOLING TRAIN, CASE 4 - FOR REFERENCE ONLY

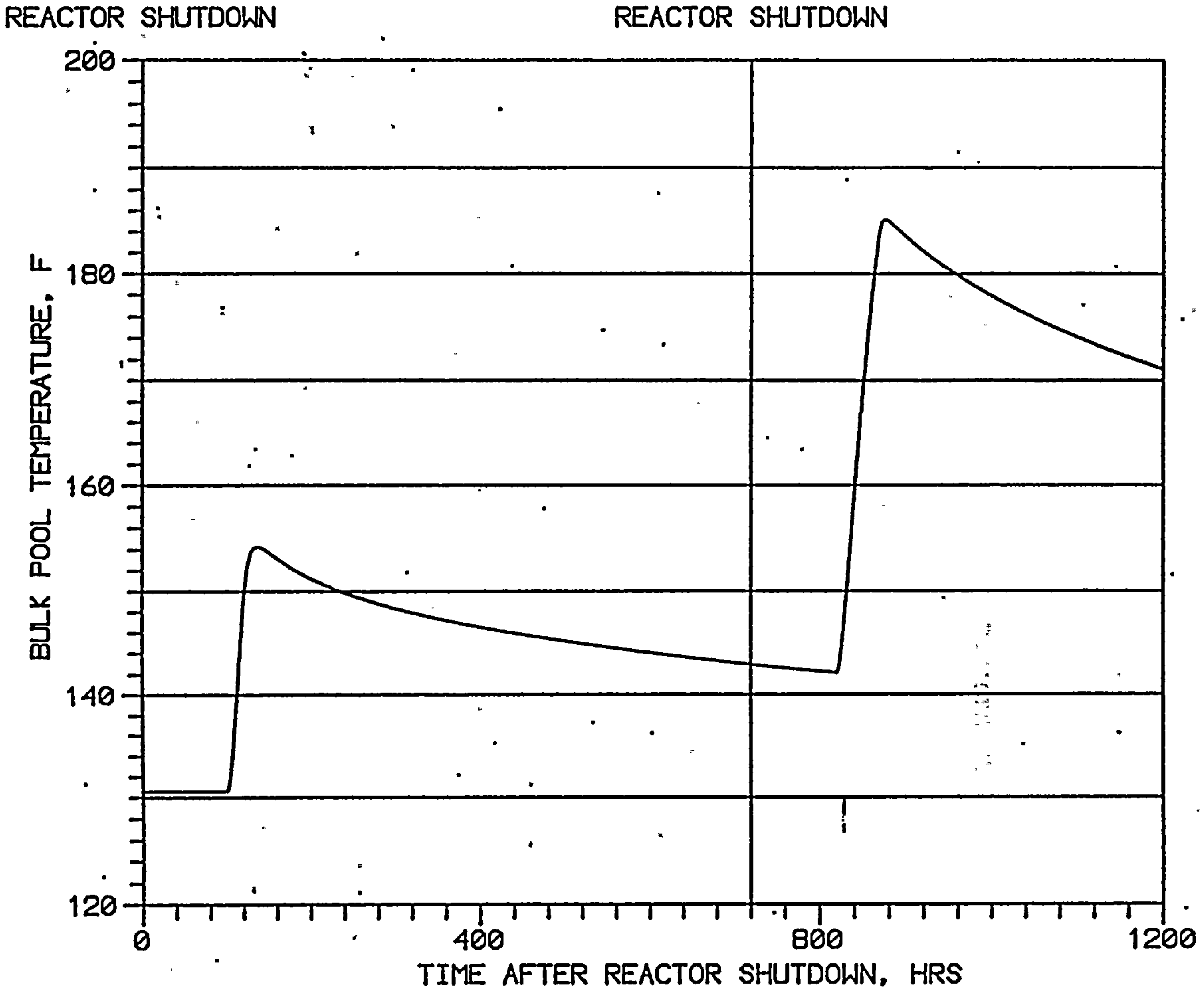

FIGURE 2.7 SFP BULK WATER TEMPERATURE PROFILE

HOLTEC INTERNATIONAL
DONALD C. COOK SPENT FUEL POOL
NORMAL DISCHARGE (80 ASSEMBLIES) 2300 GPM SFP FLOW / COOLER
ONE COOLING TRAIN, CASE 1A

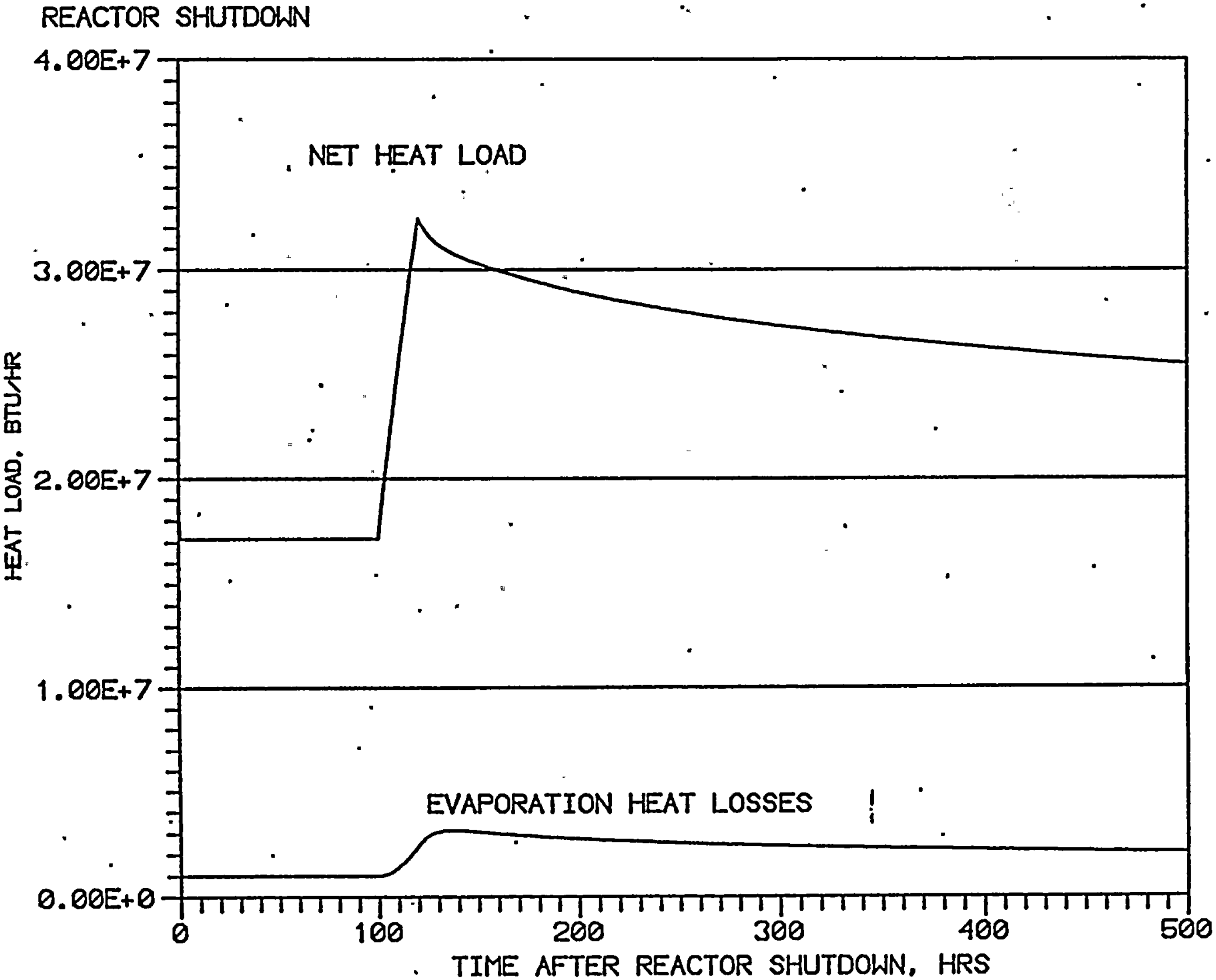

FIGURE 2.8 SFP NET DECAY HEAT LOAD AND HEAT LOSSES FOR CASE 1A

HOLTEC INTERNATIONAL
DONALD C. COOK SPENT FUEL POOL
NORMAL DISCHARGE (80 ASSEMBLIES) 2800 GPM SFP FLOW / COOLER
ONE COOLING TRAIN, CASE 1B

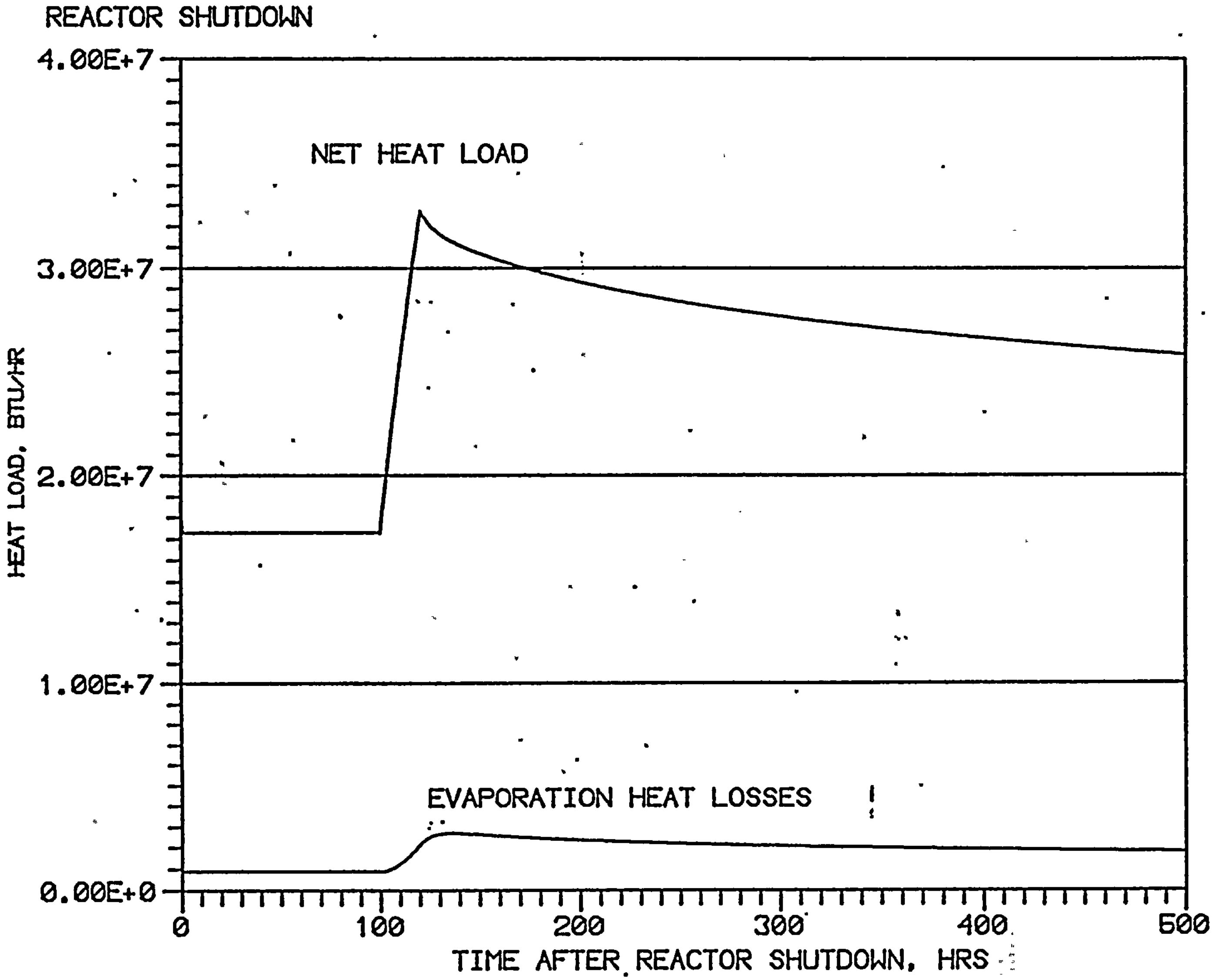

FIGURE 2.9 SFP NET DECAY HEAT LOAD AND HEAT LOSSES FOR CASE 1B

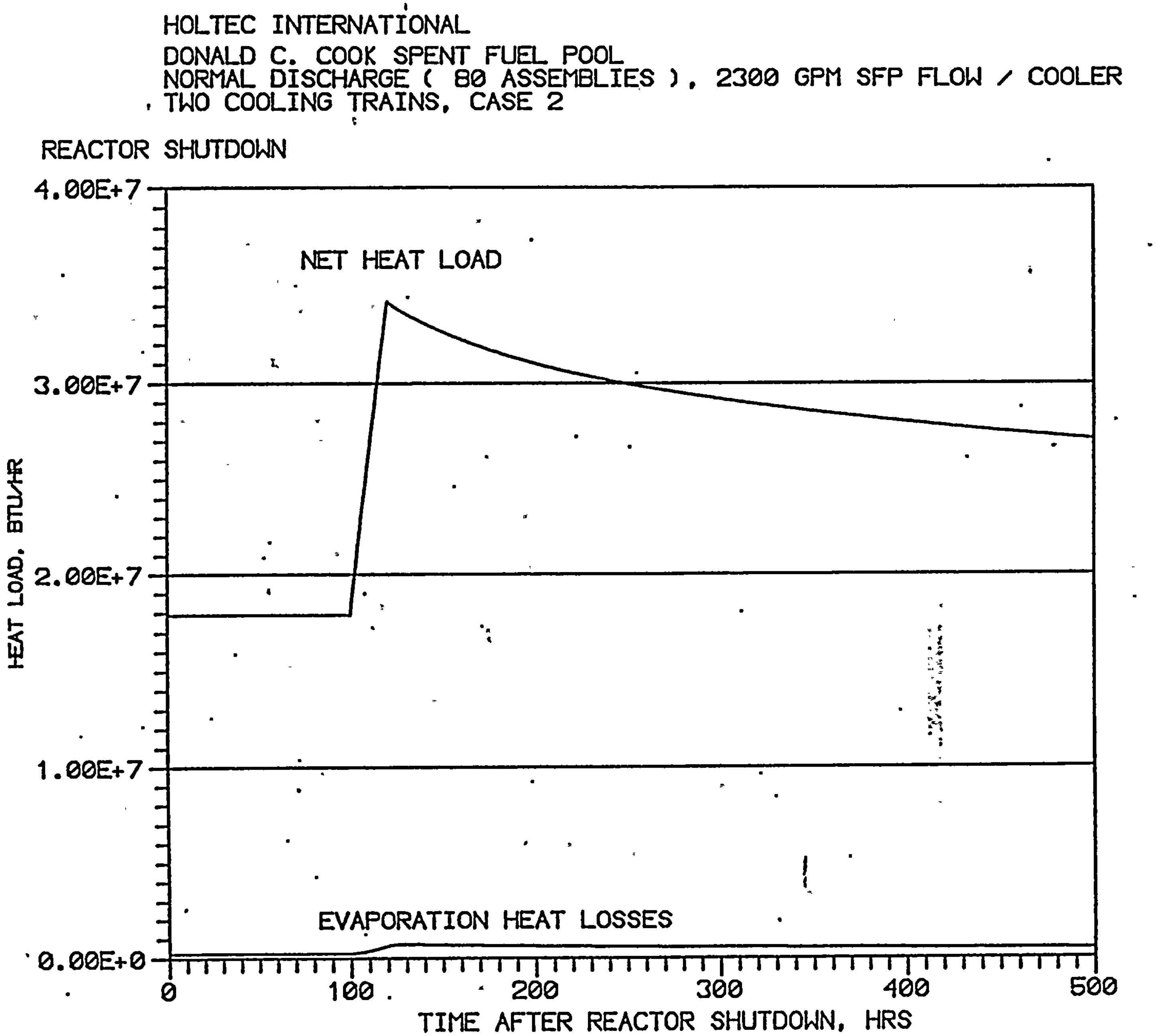

FIGURE 2.10 SFP NET DECAY HEAT LOAD AND HEAT LOSSES FOR CASE 2

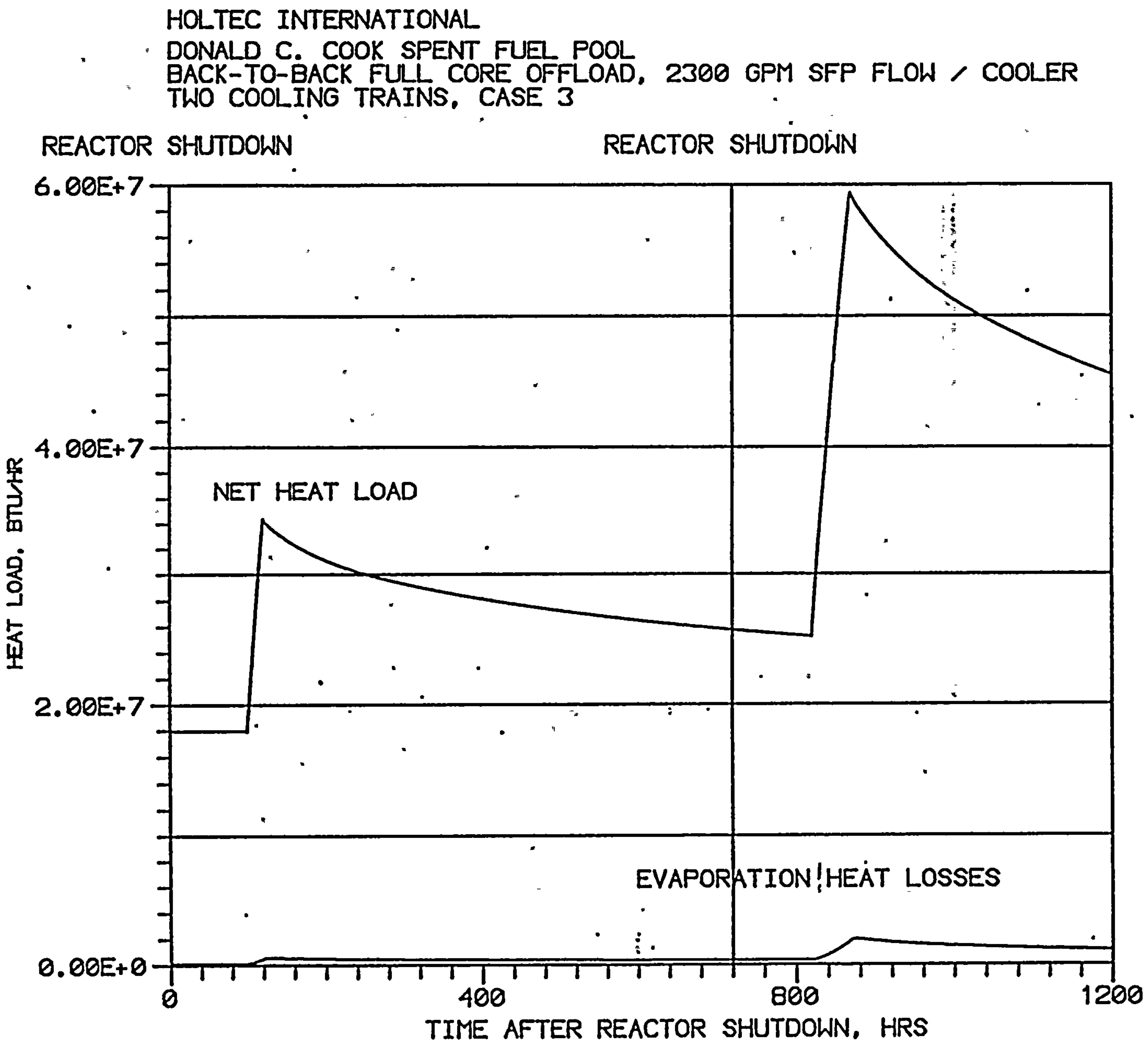

FIGURE 2.11 SFP NET DECAY HEAT LOAD AND HEAT LOSSES FOR CASE 3

HOLTEC INTERNATIONAL
DONALD C. COOK SPENT FUEL POOL
BACK-TO-BACK FULL CORE OFFLOAD, 2300 GPM SFP FLOW / COOLER
ONE COOLING TRAIN, CASE 4 - FOR REFERENCE ONLY

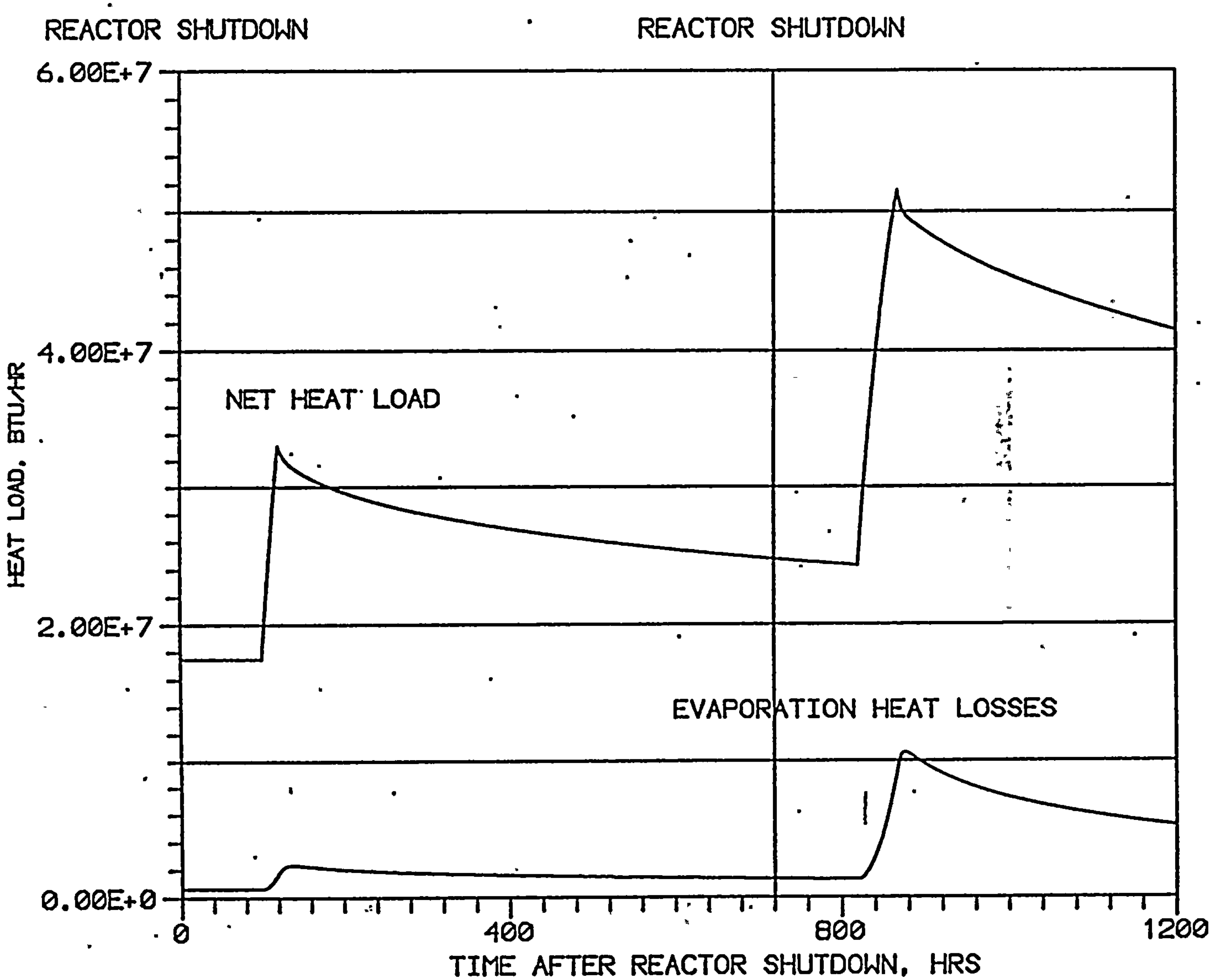

FIGURE 2.12 SFP NET DECAY HEAT LOAD AND HEAT LOSSES FOR CASE 4

3.0 EDITORIAL CHANGES TO THE RERACK LICENSING REPORT

Referring to Holtec Report HI-90488, submitted as an attachment to the 1992 licensing submittal (Amendment 169 for Unit 1 and 152 for Unit 2), the following two editorial changes are documented herein.

a. **Number of different cell types:** Figure 4.1 of the Licensing Report provided the storage pattern for Regions 1, 2 and 3 cells. While the storage cell designations in that figure are correct, the total cell counts next to the legend are not. The correct counts are as follows:

   Region 1: 503 cells
   Region 2: 1440 cells
   Region 3: 1670 cells

   Figure 4.1 (revised) is attached herein.

b. **Poison Surveillance Program:** The Boral surveillance program presented in Section 10 of the rerack licensing report [1] is somewhat unclear with respect to coupon pre-characterization and post-irradiation tests. The following paragraph is intended to clarify this item.

   All 12 coupons presently installed in the Cook Nuclear Plant fuel pool have been pre-characterized by measuring their length, width, and their thickness at discrete marked locations. In addition, their neutron transmission characteristics at discrete marked points have also been quantified using standard Holtec quality procedures for coupon testing. This pre-characterization data will serve as benchmark for future post-irradiation evaluations.

   The coupon tree will be placed in a storage cell, normally used for storing spent nuclear fuel, such that the coupons are exposed to as high a gamma field as practicable. At the time of the second discharge into the pool, number one coupon from the tree will be removed and the tree reinstalled in a storage cell, such that the coupons will, once again, continue to receive as much gamma dose as is practicable (this is evidently realized by placing the tree in a storage location which is surrounded by freshly discharged fuel).

As a minimum, the coupon removed from the tree will be measured to determine its variation in length, width, and thickness (at the pre-calibrated locations). If these physical dimensions exhibit less than 1% variation, then no further testing will be done. However, if the measured variation in any of the physical dimensions exceeds 1%, then the neutron transmission ability of the coupon (at the pre-calibrated locations) will be measured. If the post-irradiation neutron attenuation is not less than 95% of the benchmark (pre-characterized value), then no further action will be necessary. However, if the coupon fails to muster neutron attenuation acceptance capability, then it will be destructively tested to obtain a direct measure of its areal boron density by using the wet chemistry method. Should the measured boron density be found to be less than the stipulated licensing basis minimum (.030 gm/sq.cm. B-10), then the condition would warrant immediate reappraisal of criticality compliance of the storage system. The Plant's standard reporting procedures for such discrepant situations will be followed. It should be added that no plant has experienced this situation after over 200 pool years of experience with Boral.

The schedule of coupon surveillance is provided in Table 3.1.

Table 3.1 SCHEDULE OF COUPON SURVEILLANCE	
COUPON	PERIOD
1	Fall of 1994
2	1 to 2 years...(1) (2)
3	3 to 5 years...(1) (2)
4	6 to 8 years...(1) (2)
5	9 to 11 years...(1) (2)
(3)	

(1) ...after removal of Coupon No. 1.

(2) The coupon shall be removed one or two months preceding a reactor refueling (either Unit One or Two). Coupon tree will be moved to a region of high gamma flux during the reactor refueling outage (i.e., surrounded by freshly discharged fuel) when a coupon has been removed from the pool.

(3) Repeat the test every five years for the remaining duration of wet storage in the Donald C. Cook spent fuel pool.

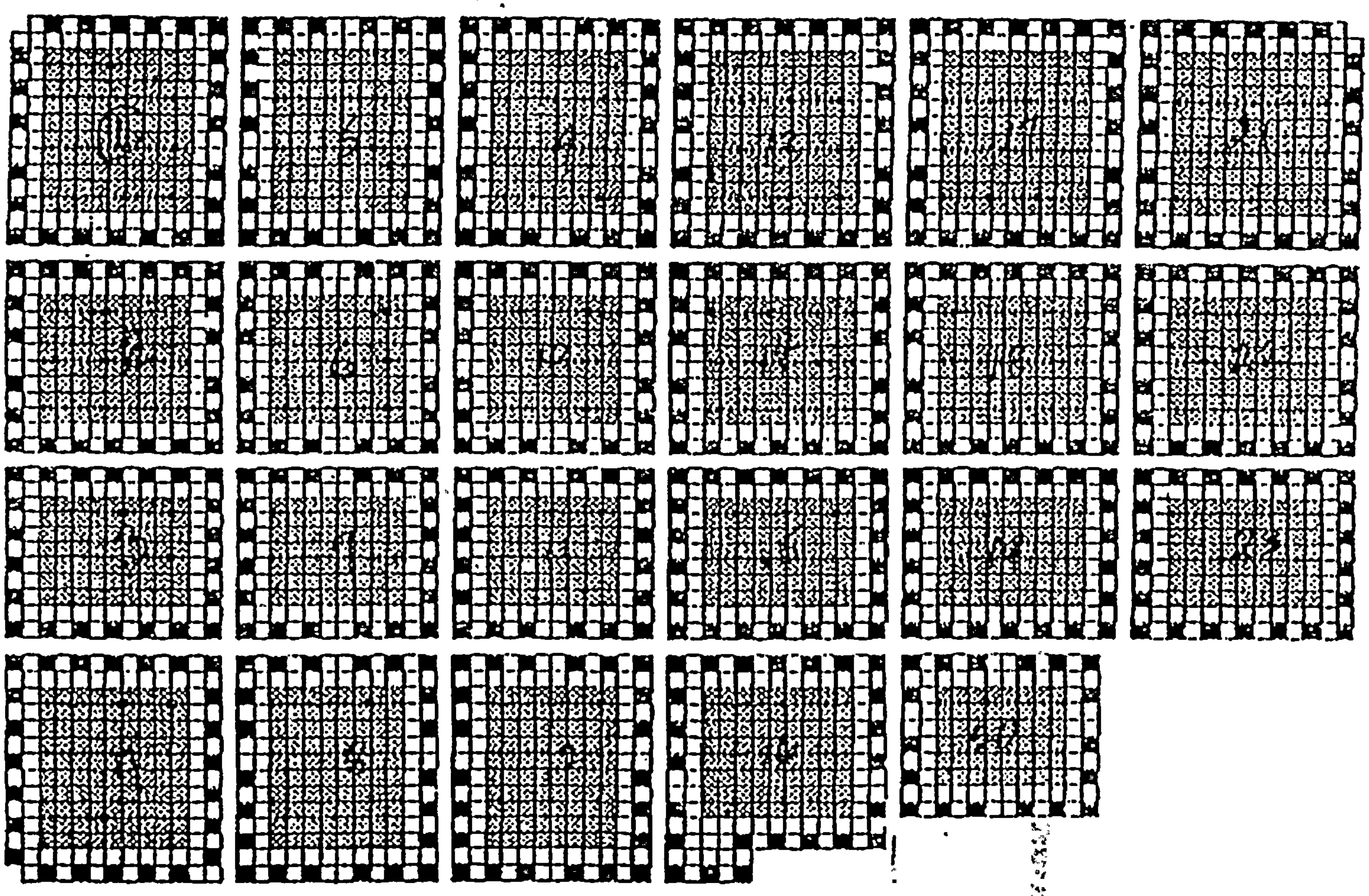

Fig. 4-1 NORMAL STORAGE PATTERN (MIXED THREE ZONE)

~~504~~ 503 REGION 1 CELLS

~~1116~~ 1440 REGION 2 CELLS

~~1694~~ 1670 REGION 3 CELLS

Revised 5/5/94

4.0 REFERENCES

[1] Letter from E.E. Fitzpatrick to T.E. Mulrey, USNRC, AEP: NRC: 1146, dated July 26, 1991 and attachments (includes Holtec Licensing Report HI-90488 as one of the attachments).