5. THERMAL-HYDRAULIC CONSIDERATIONS

A central objective in the design of the high-density fuel rack is to ensure adequate cooling of the fuel assembly cladding. In the following, a brief synopsis of the design basis, the method of analysis, and computed results is given.

5.1 Decay Heat Calculations for the Spent Fuel

This report section covers requirement III.1.5(2) of the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" issued on April 14, 1978. This requirement states that calculations for the amount of thermal energy removed by the spent fuel cooling system shall be made in accordance with Branch Technical Position APCSB 9-2 "Residual Decay Energy for Light Water Reactors for Long Term Cooling"[2]. The calculations contained herein have been made in accordance with this requirements.

5.1.1 Basis: 2

The Quad Cities 1 and 2 reactors are rated at 2511 Megawatt-Thermal (MWT) each. The core contains 724 fuel assemblies. Thus, the average operating power per fuel assembly, P_o, is 3.468 MW. The fuel assemblies are removed from the reactor after a nominal burn-up of 25000 Megawatt-days per short ton of uranium (MWD/STU). The fuel discharge can be made in one of the following two modes:

(i) Normal discharge - Mode (i)
(ii) Full Core discharge - Mode (ii)

As shown in Table 1.1 of Section 1, the average fuel assembly removal batch size for Mode (i) is 200 fuel assemblies. However, the computations are performed for a batch as large as 240 fuel assemblies. The fuel transfer begins after 100 hours of cool-off time in the reactor (time after shut down). It is assumed that the time period of discharge of this batch is 2 days.

Mode (ii) corresponds to a full core discharge (724 assemblies). It is assumed that the total time period for the discharge of one full core is 6 days after 100 hours of shut down time in the reactor). The discharge rate to the pool is assumed to be continuous and uniform.

The heat dissipation from each pool is accomplished by two independent fuel pool cooler loops, each equipped with a pump rated at 700 gpm. In addition, the Residual Heat Removal (RHR) heat exchangers may be used in conjunction with the fuel pool coolers to boost the heat removal rate. For each unit, there are two RHR heat exchangers supplied by four pumps, three of which can deliver 14500 gpm at 360' head. Despite the large potential capacity, it has been assumed that only 1000 gpm of this flow rate is available for the fuel pool, through one six inch pipe line.

In the following, all relevant performance data for the spent fuel pool and RHR heat exchangers is given.

a. Spent Fuel Pool Heat Exchanger: 2

TEMA type 21-197 BEU, 1020 sq.ft. surface on 196 U-tubes; 5/8" diameter x 18 BWG arranged on 0.875" triangular pitch. Postulated fouling for both tube and shellside surfaces is 0.0005 sq.ft-^{o}F-Hr/BTU (each surface). Shellside (cooling) and tubeside (pool water) flow rates are 800,000 and 350,000 lbs/hr. respectively. The corresponding value of the reduced thermal flux (NTU) for fully fouled condition is 0.933; and the temperature efficiency is 0.55.

b. RHR Heat Exchanger:

TEMA type 63-288 CET, 11000 sq.ft. overall surface on 2415 tubes, 3/4" diameter x 18 BWG arranged on 0.9375" triangular pitch. Postulated fouling for the shellside of the tube surface is 0.0005 sq.ft.-^{o}F-Hr./BTU and that for the tubeside (river water) is .002 sq.ft.-^{o}F-Hr./BTU.

The shellside and tubeside design flow rates are 5.35×10^6 lbs/hr. and 3.5×10^6 lb/hr., respectively. The corresponding value of NTU is 0.745; and the temperature efficiency is 0.385.

The above data enables complete characterization of the thermal performance of the heat exchangers.

Reference[2] is utilized to compute the heat dissipation requirements in the pool. The total decay power consists of "fission products decay" and "heavy element decay." Total decay power P for a fuel assembly is given as a linear function of P_o and an exponentional function of t_o and t_s.

ie: $P = P_o \, f(t_o, t_s)$

where

P = linear function of P_o

P_o = average operating power per fuel assembly

t_o = cumulative exposure time of the fuel assembly in the reactor

t_s = Time elapsed since reactor shutdown

The uncertainty factor K, which occurs in the functional relationship $f(t_o, t_s)$ is set equal to 0.1 for $t_s > 10^7$ sec in the interest of conservatism. Furthermore, the operating power P_o is taken equal to the rated power, even though the reactor may be operating at a fraction of its total power during most of the period of exposure of the batch of fuel assemblies. Finally, the computations and results reported here are based on the discharge in the year 2005

(ref. Table 1.1). This is when the inventory of fuel in the pool will be at its maximum resulting in an upper bound on the computed decay heat rate.

In the past, Quad Cities reactors have operated on what is commonly referred to as "18 month cycle." Quite often, system planning requires extended reactor coastdown operation (sometimes to 40% of rated power) after the end of full power reactivity (19000 MWD/STU) has been reached. The batch average discharge burn-up of current fuel batches is approximately 25000 MWD/STU. In the future, due to present lack of spent fuel reprocessing in the U.S., it is conceivable that the average discharge exposure can approach 30,000 MWD/STU due to higher initial enrichments and longer coastdowns. A longer coastdown period implies a greater value of t_o in the foregoing equation, it also implies a smaller value of P_o. It can be shown that an exposure period, t_o, equal to 4.5 years (3-18 month refueling cycles) along with the rated reactor power produces an upper bound on the value of P. This is due to the fact that f (to,ts) is a weak monotonically increasing function of t_o. Hence, the reactor operating time is assumed to be 4.5 years (t_o=1.42x10^8 secs). 2

Having determined the heat dissipation rate, the next task is to evaluate the time temperature history of the pool water. Table 5.1.1 identifies the loading cases examined. The pool bulk temperature time history is determined using the first law of thermodynamics (conservation of heat). The system to be analyzed is shown in Figure 5.1.1.

A number of simplifying assumptions are made to render the analysis conservative. The principal ones are:

1. The cooling water temperature in the fuel pool cooler and the RHR heat exchangers are based on the maximum postulated values given in the FSAR.

2. The heat exchangers are assumed to have maximum fouling. Thus, the temperature effectiveness, S, for the heat exchangers utilized in the analysis are the lowest postulated values: S= 0.52 for fuel pool coolers, 0.385 for RHR heat exchangers. S is calculated from FSAR and heat exchanger technical data sheets.

3. No heat loss is assumed to take place through the concrete floor.

4. No credit is taken for the improvement in the film coefficients of the heat exchangers as the operating temperature rises. Thus, the film coefficient used in the computations are lower bounds.

5. No credit is taken for evaporation of the pool water.

The basic energy conservation relationship for the pool heat exchanger system yields: 2

$$C_t \frac{dt}{d\tau} = Q_1 - Q_2 - Q_3 \qquad (5.1.2)$$

where

C_t: Thermal capacity of stored water in the pool.
t: Temperature of pool water at time, τ

Q_1: Heat generation rate due to stored fuel assemblies in the pool. Q_1 is a known function of time, τ from the preceding section.

Q_2: Heat removed in the two fuel pool coolers.

Q_3: Heat removed in the RHR heat exchanger (Q_3=0 if RHR is not used).

TABLE 5.1.1

LIST OF CASES ANALYZED

Case No.	Condition	No. of fuel assemblies N	No. of spent fuel pool HXS	No. of RHR's in-service	Total Time to transfer fuel into the pool t_h, hrs.	Cool off time before transfer begins, hrs.
1	Normal discharge with oversize batch	240	2	0	48	100
2	Same as One	240	2	1	48	100
3	Normal discharge	200	2	0	48	100
4	Normal discharge	200	2	1	48	100
5	Full Core Discharge	724	2	1	144	100

The pools for Unit 1 and 2 in the Quad Cities installations have total water inventory of 44887 and 44471 cubic feet respectively when all racks are in place in the pools and every storage location is occupied.

5.1.2 Decay Heat Calculation Results:

The calculations were performed for the Quad Cities Unit 2 pool disregarding the additional thermal capacity and cooling system available in the other pool. The use of the lower water inventory of the Unit 2 pool thus is the bounding case.

For a specified coolant inlet temperature and flow rate, the quantities Q_2 and Q_3 are shown to be linear function of t in a recent paper by Singh[3]. As stated earlier, Q_1, is an exponential function of τ . Thus Equation (5.1.2) can be integrated to determine t directly as a function of τ . The results are plotted in Figures (5.1.2) - Figures (5.1.11) and show that the pool water never approaches the boiling point under the most adverse conditions. These figures also give Q_1 as a function of τ . Two plots are generated for each case. The first plot for each shows temperature and power generation for a period extending from $\tau = 0 \longrightarrow \tau = 2\tau_n$ where τ_n is the total time of fuel transfer. The second plot shows the same quantities over a long period. The long-term plots are produced to indicate the required operating time for the heat exchangers. Summarized results are given in Table 5.1.2. 2

Finally, computations are made to determine the time interval to boiling after all heat dissipation paths are lost. Computations are made for each case under the following two assumptions:

(i) All cooling sources lost at the instant pool bulk temperature reaches the maximum value

(ii) All cooling paths lost at the instant the heat dissipation power reaches its maximum value in the pool.

Results are summarized in Table 5.1.3. Table 5.1.3 gives the bulk boiling vaporization rate for both cases at the instant the boiling commences. This rate will decrease with time due to reduced heat emission from the fuel.

5.2 Thermal-Hydraulics Analyses for Spent Fuel Cooling

2

This report section covers requirement III.1.5(3) of the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" issued on April 14, 1978. Conservative methods have been used to calculate the maximum fuel cladding temperature as required therein. Also, it has been determined that nucleate boiling or voiding of coolant on the surface of the fuel rods does not occur.

5.2.1 Basis:

In order to determine an upper bound on the maximum fuel cladding temperature, a series of conservative assumptions are made. The most important assumptions are listed below:

a. As stated above, the fuel pool will contain spent fuel with varying "time-after-shutdown" (t_s). Since the heat emission falls off rapidly with increasing t_s, it is obviously conservative to assume that all fuel assemblies are fresh (t_s = 100 hours), and they all have had 4.5 years of operating time in the reactor. The heat emission rate of each fuel assembly is assumed to be equal.[2]

b. As shown in Figures 2.1 and 2.2 in Section 2, the modules occupy an irregular floor space in the pool. For purposes of the hydrothermal analysis, a circle circumscribing the

TABLE 5.1.2

MAXIMUM POOL BULK TEMPERATURE t, COINCIDENT TOTAL POWER Q_1 and

COINCIDENT SPECIFIC POWER FOR THE HOTTEST ASSEMBLY

Case No.	No. of Assemblies	n Time to transfer fuel into pool, hrs.	Maximum pool bulk temp. ^{o}F	Coincident time (since initiation of fuel transfer, hrs.	Coincident specific power q, BTU/sec.	$Q_1 x 10^{-6}$ BTU/hour
1	240	48	134.6	64	10.24	10.85
2	240	48	121.2	58	10.39	10.99
3	200	48	130.6	64	10.24	9.37
4	200	48	118.5	58	10.39	9.48
5	724	144	145.8	150	8.72	24.89

TABLE 5.1.3

TIME (Hrs) TO BOILING AND BOILING VAPORIZATION RATE
FROM THE INSTANT ALL COOLING IS LOST

Case No.	CONDITION 1 Loss of Cooling at maximum pool bulk temperature		CONDITION 2 Loss of Cooling at maximum power discharge rate	
	Time (Hrs)	Vap. Rate lb./hr.	Time (Hrs)	Vap. Rate lb./hr.
1	20.3	10759	20.4	11038
2	23.7	10723	24.5	10955
3	24.8	9203	24.7	9461
4	28.2	9224	29.1	9389
5	7.7	25147	7.64	25384

actual rack floor space is drawn. It is further assumed that the cylinder with this circle as its base is packed with fuel assemblies at the nominal pitch of 6.22 inches (see Figure 5.2.1).

c. The downcomer space around the rack module group varies, as shown in Figure 5.2.1. The nominal downcomer gap (9 inches) available in the pool is assumed to be the total gap available around the idealized cylindrical rack; thus, the maximum resistance to downward flow is incorporated into the analysis.

d. No downcomer flow is assumed to exist between the rack modules.

In this manner, a conservative idealized model for the rack assemblage is devised. The water flow is axisymmetric about the vertical axis of the circular rack assemblage, and thus, the flow is two-dimensional (axisymmetric three-dimensional). The governing equation to characterize the flow field in the pool can now be written. The resulting integral equation can be solved for the lower plenum velocity field (in the radial direction) and axial velocity (in-cell velocity field), by using the method of collocation. It should be added here that the hydrodynamic loss coefficients which enter into the formulation of the integral equation are also taken from well-recognized sources[4] and wherever discrepancies in reported values exist, the conservative values are consistently used.

After the axial velocity field is evaluated, it is a straightforward matter to compute the fuel assembly cladding temperature. The knowledge of the overall flow field enables pinpointing the storage location with the minimum axial flow (i.e., maximum water outlet temperature). This is called the most "choked" location. It is recognized that some storage locations, where rack module supports are located, have some additional hydraulic resistance not encountered in

other cells. In order to find an upper bound on the temperature in such a cell, it is assumed that it is located at the most "choked" location. Knowing the global plenum velocity field, the revised axial flow through this choked cell can be calculated by solving the Bernoulli's equation for the flow circuit through this cell. Thus, an absolute upper bound on the water exit temperature and maximum fuel cladding temperature is obtained. It is believed that in view of the preceding assumption, the temperatures calculated in this manner overestimate the temperature rise that will actually be obtained in the pool.

The maximum pool bulk temperature t is computed in Section 5.1.3 and reported in Table 5.1.2. The corresponding average power output from the hottest fuel assembly, q is also reported in that table. The maximum radial peaking factor, ranges from 1.6 to 1.8 for Quad Cities installations. Thus, it is conservative to assume that the maximum specific power of a fuel assembly is given by

$$q_A = q\,\alpha_r$$

where $\alpha_r = 1.8$

The maximum temperature rise of pool water in the most disadvantageously placed fuel assembly is given in Table 5.2.1 for all loading cases. Having determined the maximum "local" water temperature in the pool, it is now possible to determine the maximum fuel cladding temperature. It is conservatively assumed that the total peaking factor α_T is 3.1. Thus, a fuel rod can produce 3.1 times the average heat emission rate over a small length. The axial heat dissipation in a rod is known to reach a maximum in the central region; and taper off at its two extremities. For the sake of added conservatism it is assumed that the peak heat emission occurs at the top where

2

TABLE 5.2.1

MAXIMUM LOCAL POOL WATER TEMPERATURE AND LOCAL FUEL

CLADDING TEMPERATURE

Case No.	Max. Local Pool Water Temperature ^{o}F	Maximum Coincident Local Cladding Temperature ^{o}F	Case Identified
1	157.8	183.6	240 Assemblies Cooling Mode A
2	144.7	170.8	240 Assemblies Cooling Mode B
3	153.8	179.6	200 Assemblies Cooling Mode A
4	142	168.1	200 Assemblies Cooling Mode B
5	166.6	189.0	724 Assemblies Cooling Mode B

* Cooling Mode A mean only two fuel pool Hxs working.

Cooling Mode B means two fuel pool and Portion of 1 RHR working.

2

TABLE 5.2.2

POOL AND MAXIMUM CLADDING TEMPERATURE AT THE INSTANCE FUEL ASSEMBLY TRANSFER BEGINS

Case No.	Cladding Temp. °F	Coincident Pool Temp, °F	
		Bulk	Local
1	178.3	110.5	133.7
2	173.3	105.2	128.7
3	178.3	110.5	133.7
4	173.3	105.2	128.7
5	170.6	105.2	126.0

the local water temperature also reaches its maximum. Furthermore, no credit is taken for axial conduction of heat along the rod. The highly conservative model thus constructed leads to simple algebraic equations which directly give the maximum local cladding temperature, t_c.

5.2.2 Results

Table 5.2.1 gives the maximum local cladding temperature, t_c, at the instant the pool bulk temperature has attained its maximum value. It is quite possible, however, that the peak cladding temperature occurs at the instant of maximum value of q_A, i.e., at the instant when the fuel assembly is first placed in a storage location. Table 5.2.2 gives the maximum local cladding temperature at =0. It is to be noted that there are wide margins to local boiling in all cases. The local boiling temperature near the top of the fuel cladding is 240°F. Furthermore, the cladding temperature must be somewhat higher than the boiling temperature to initiate and sustain nucleate boiling. The above considerations indicate that a comfortable margin against the initiation of localized boiling exists in all cases.

REFERENCES TO SECTION 5

1. FSAR, Quad Cities, Section 10, Auxiliary and Emergency Systems.

2. U.S. Nuclear Regulatory Commission, Standard Review Plan, Branch Technical Position, APCSB 9-2, Rev. 1, November 1975.

3. Journal of Heat Transfer, Transactions of the ASME (c. 81), "Some Fundamental Relationships for Tubular Heat Exchanger Thermal Performance," K.P. Singh.

4. General Electric Corporation, R&D Data Books, "Heat Transfer and Fluid Flow," 1974 and updates.

2

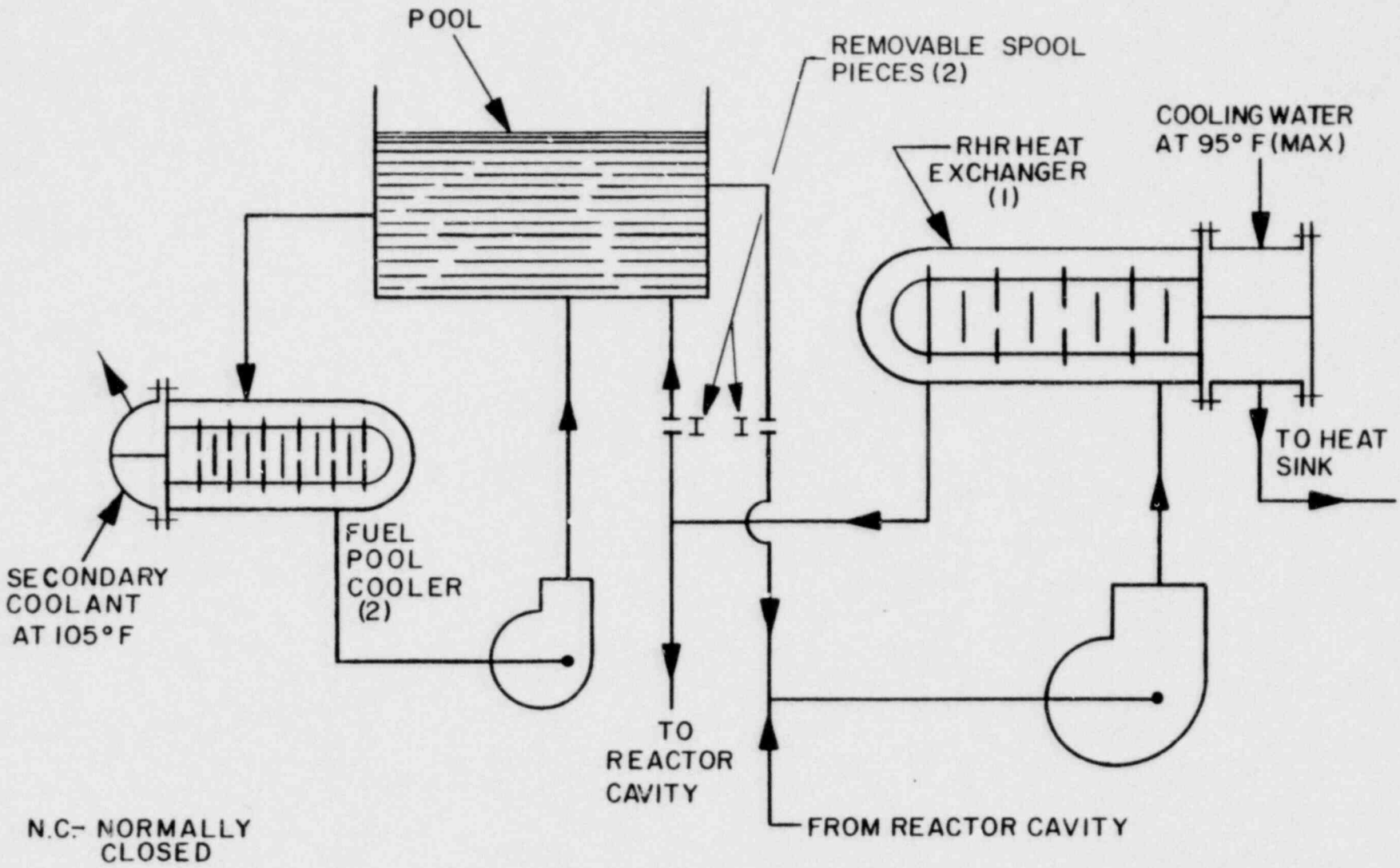

FIG 5.1.1 IDEALIZED POOL COOLING SYSTEM

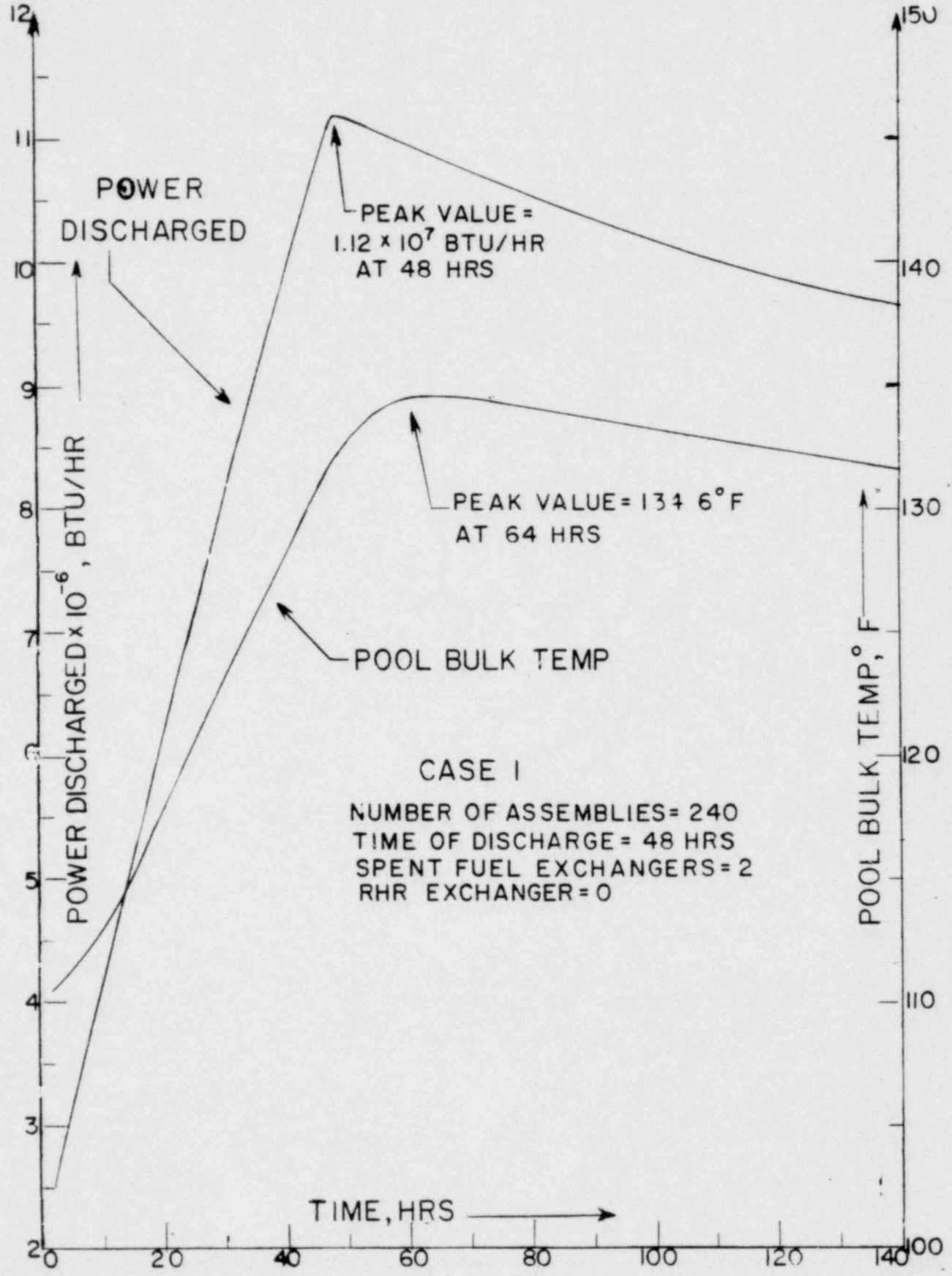

FIG 5.1.2 ; NORMAL DISCHARGE, 240 ASSEMBLIES

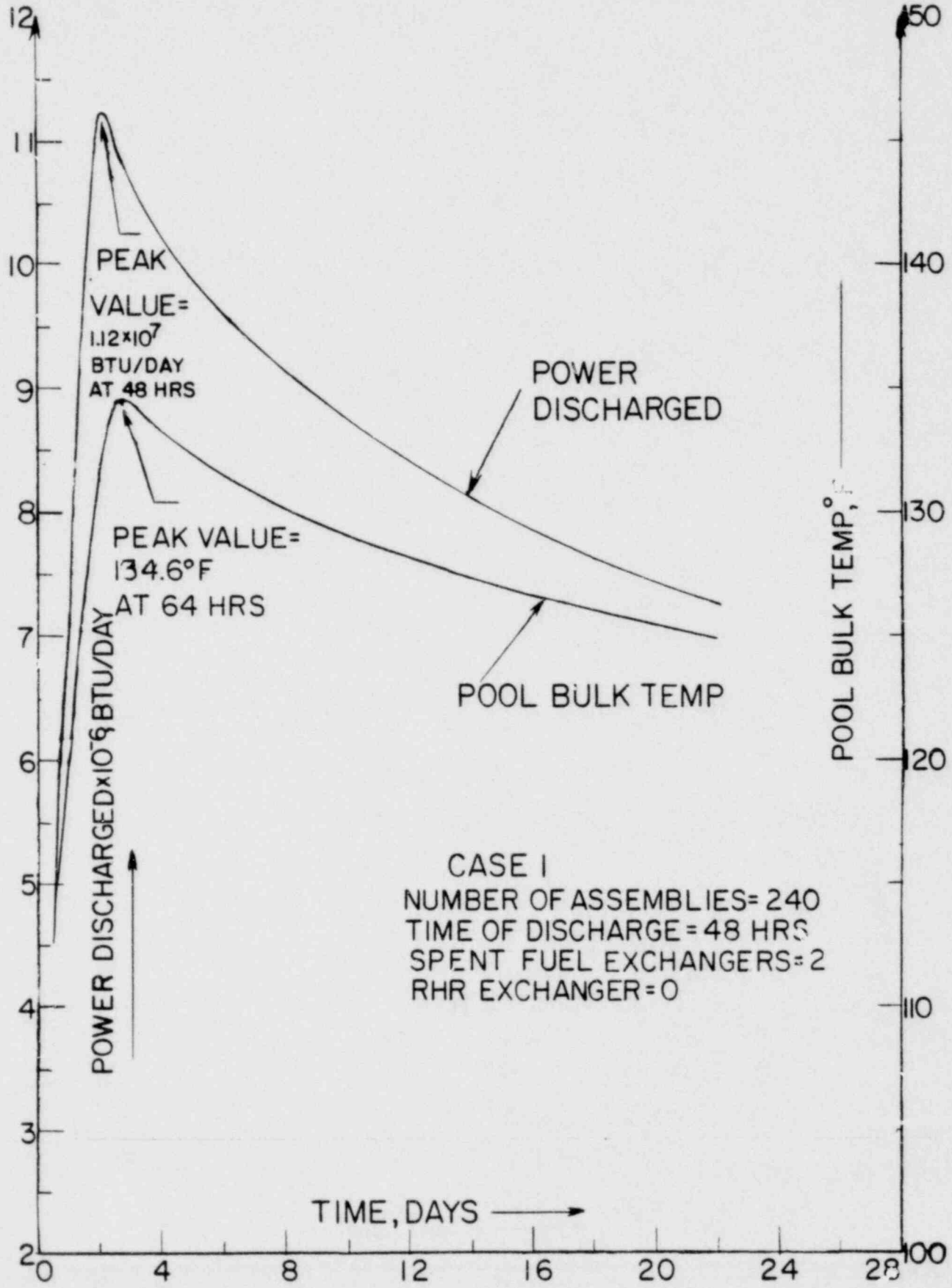

FIG 5.1.3; NORMAL DISCHARGE, 240 ASSEMBLIES

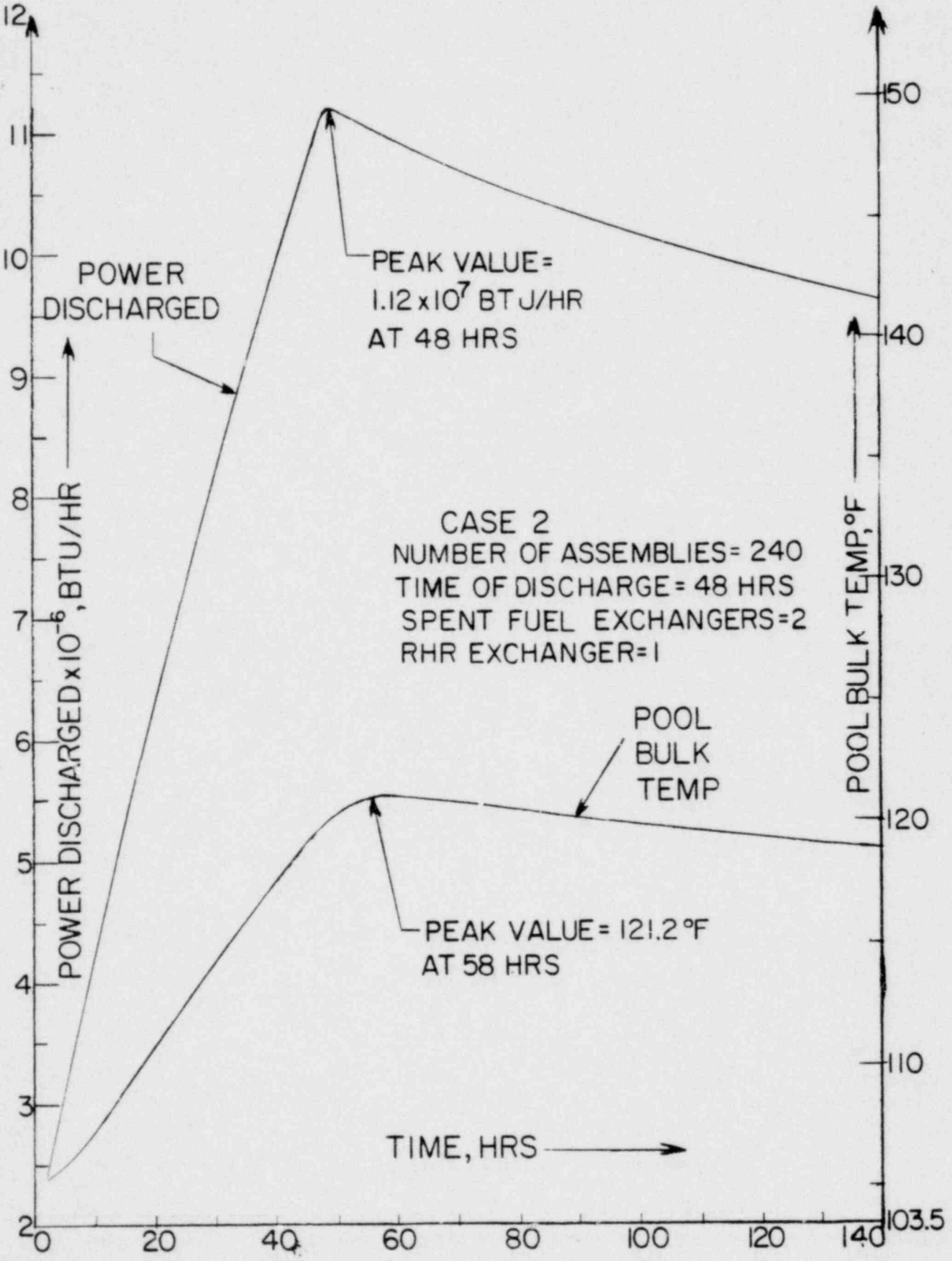

FIG 5.1.4; NORMAL DISCHARGE, 240 ASSEMBLIES

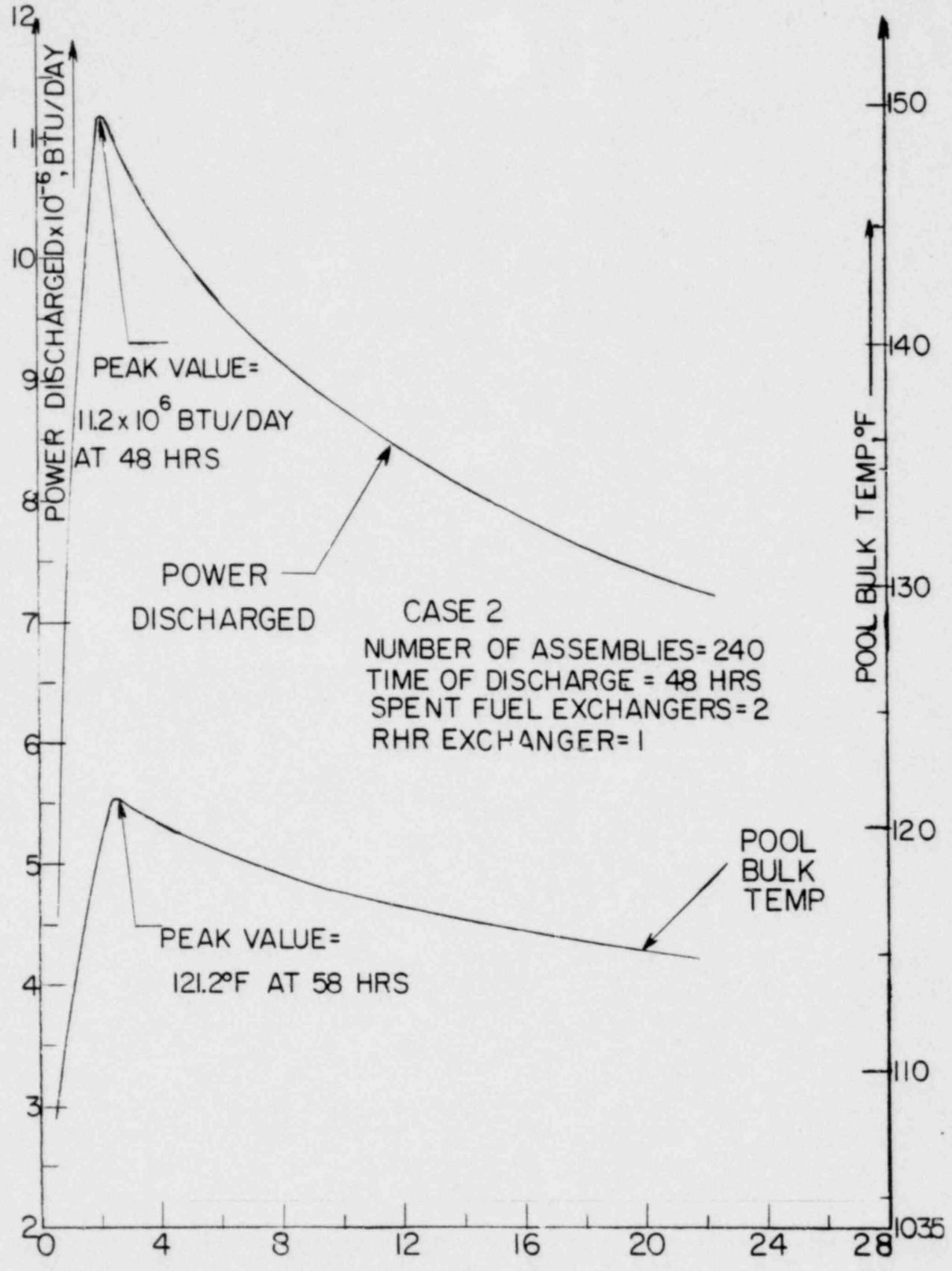

FIG 5.1.5; NORMAL DISCHARGE, 240 ASSEMBLIES

TIME, DAYS ⟶

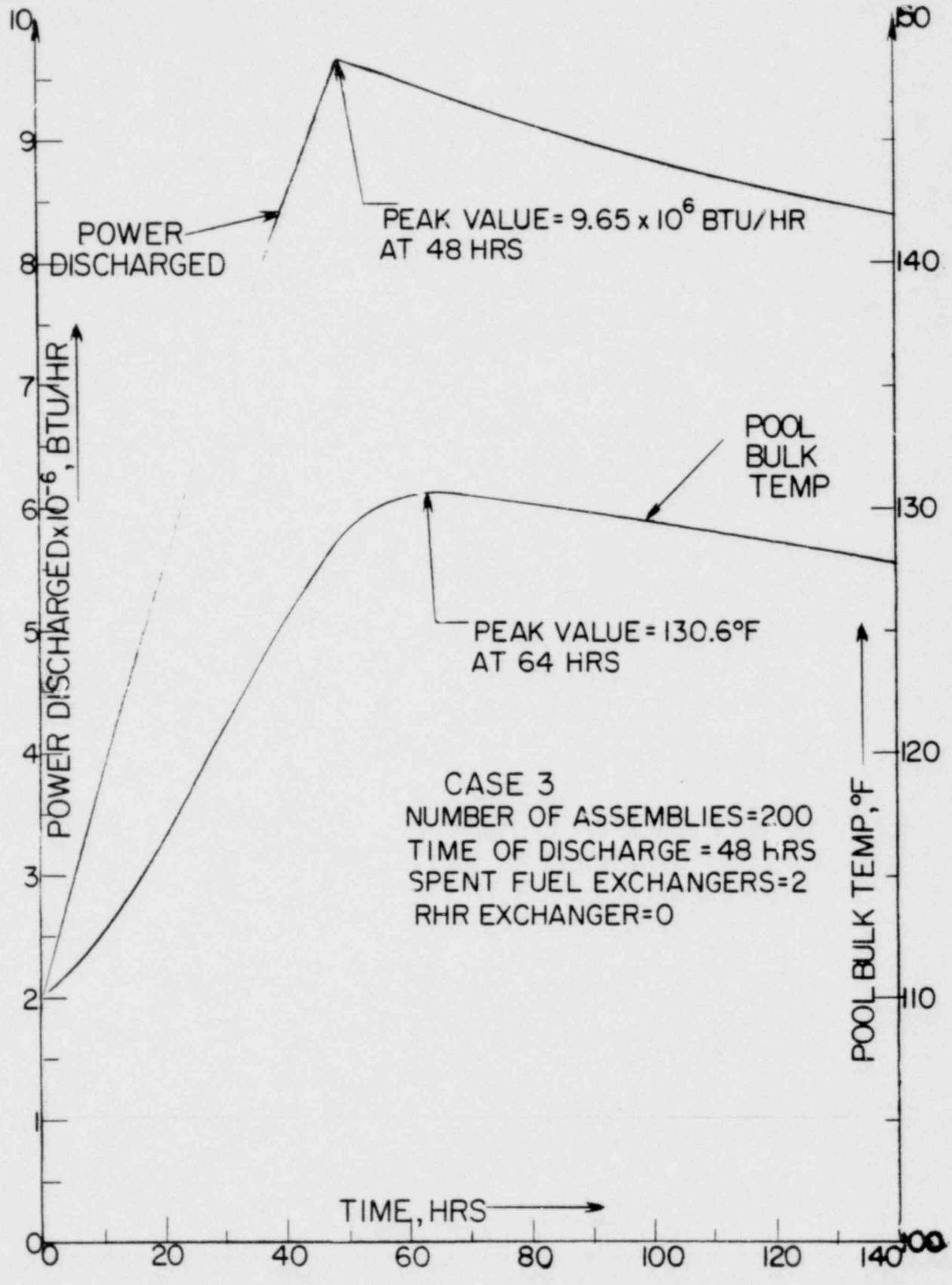

FIG 5.1.6; NORMAL DISCHARGE, 200 ASSEMBLIES

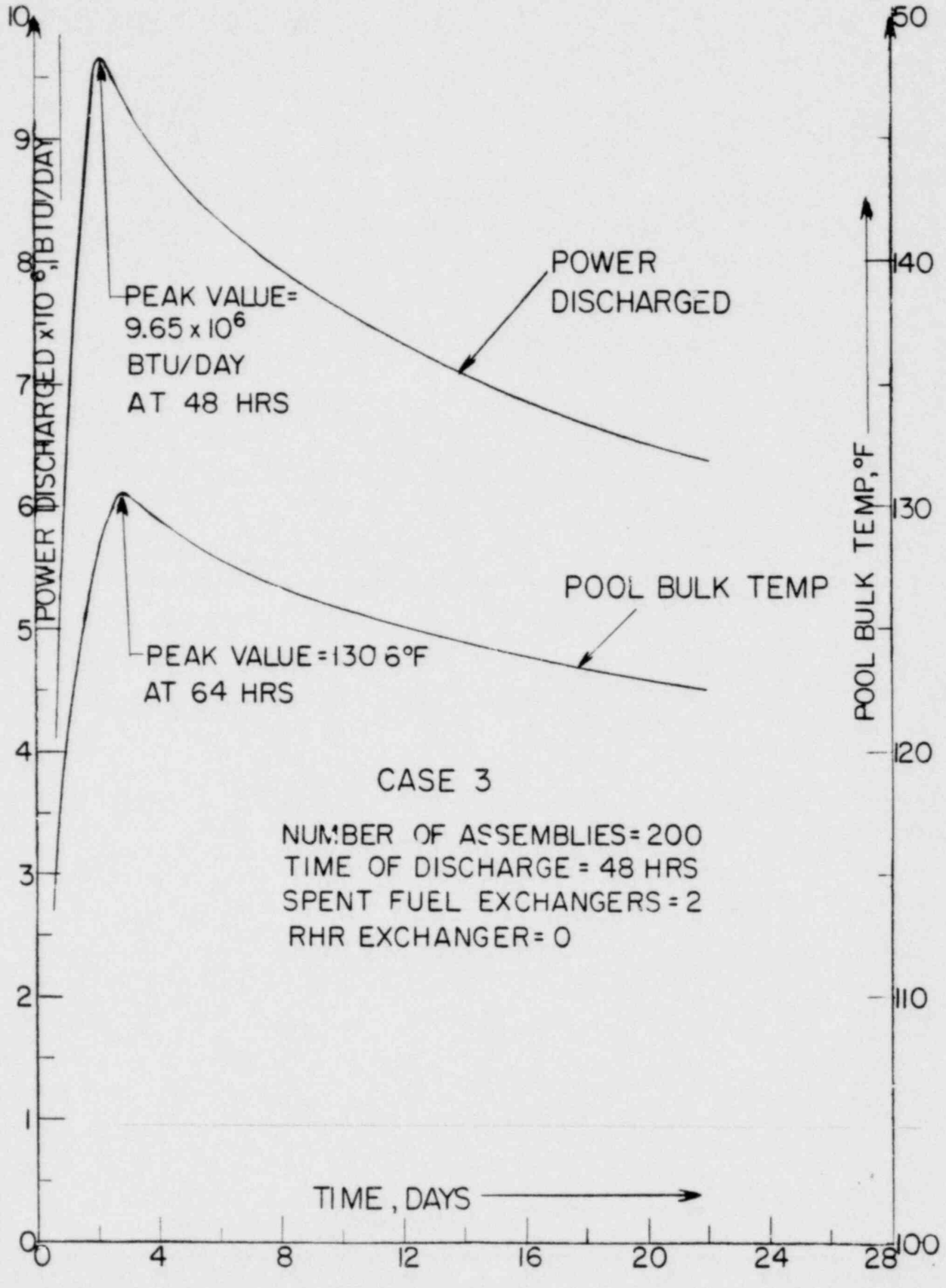

FIG 5.1.7; NORMAL DISCHARGE, 200 ASSEMBLIES

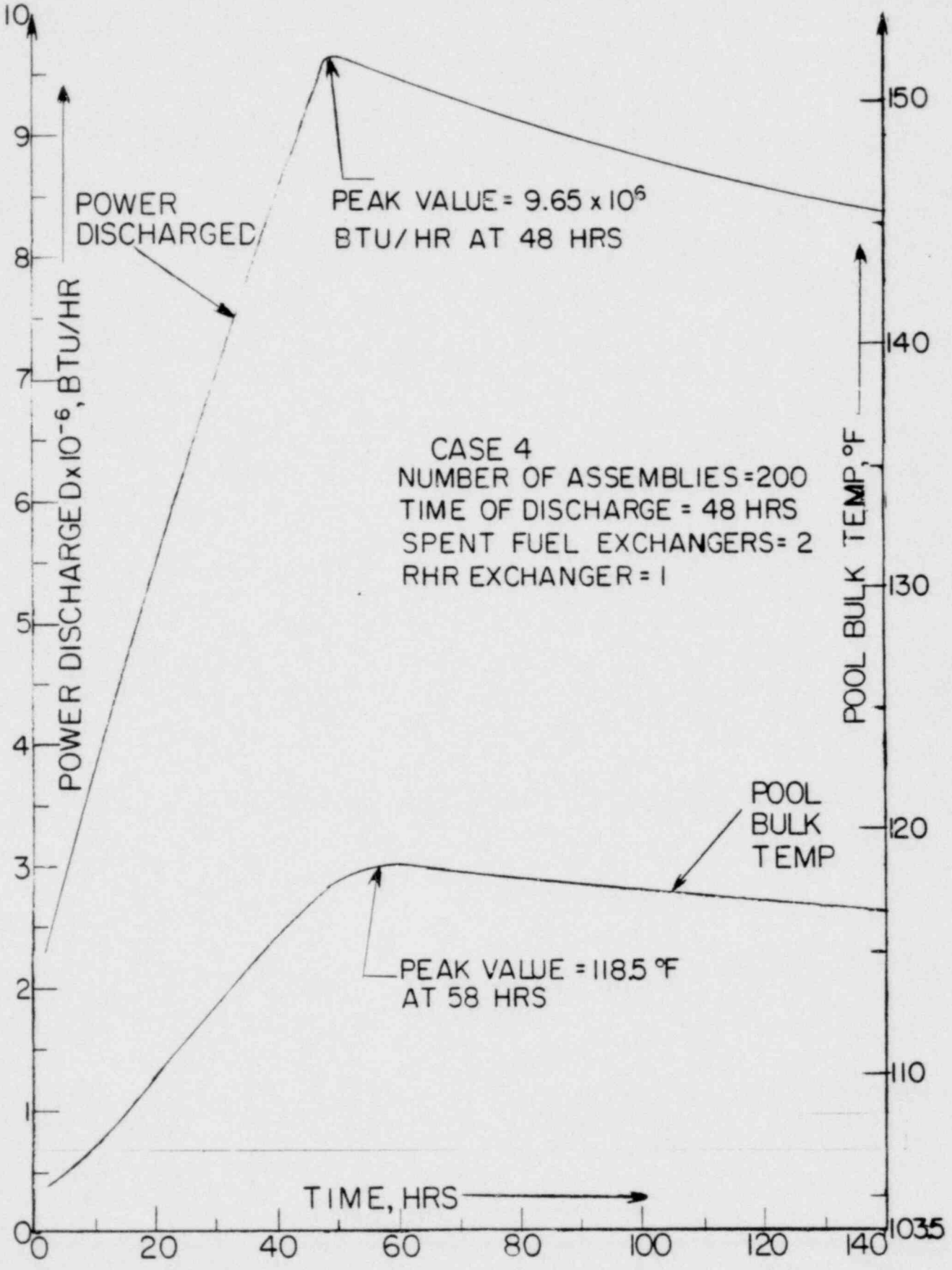

FIG 5.1.8; NORMAL DISCHARGE, 200 ASSEMBLIES

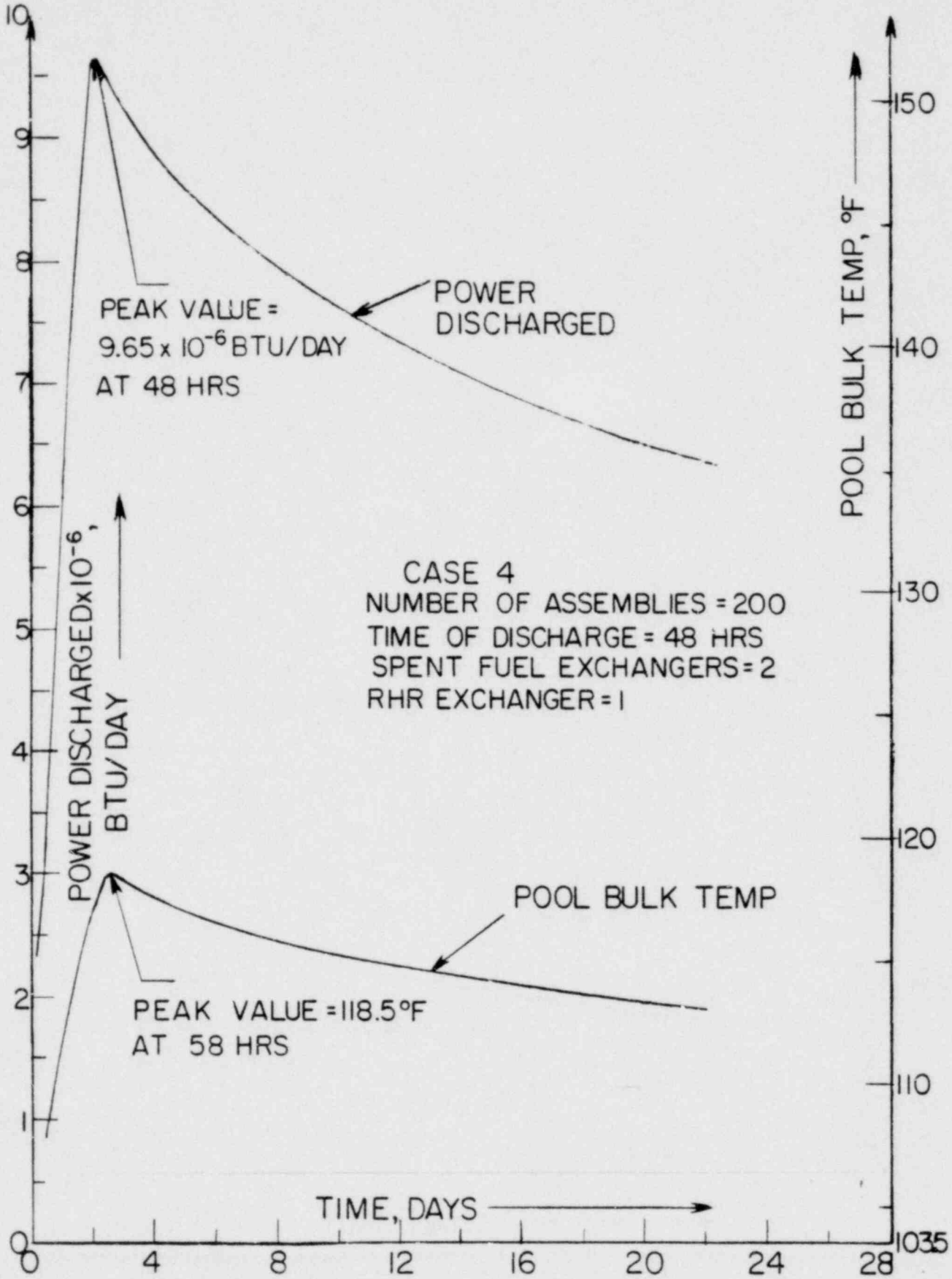

FIG 5.1.9
NORMAL DISCHARGE, 200 ASSEMBLIES

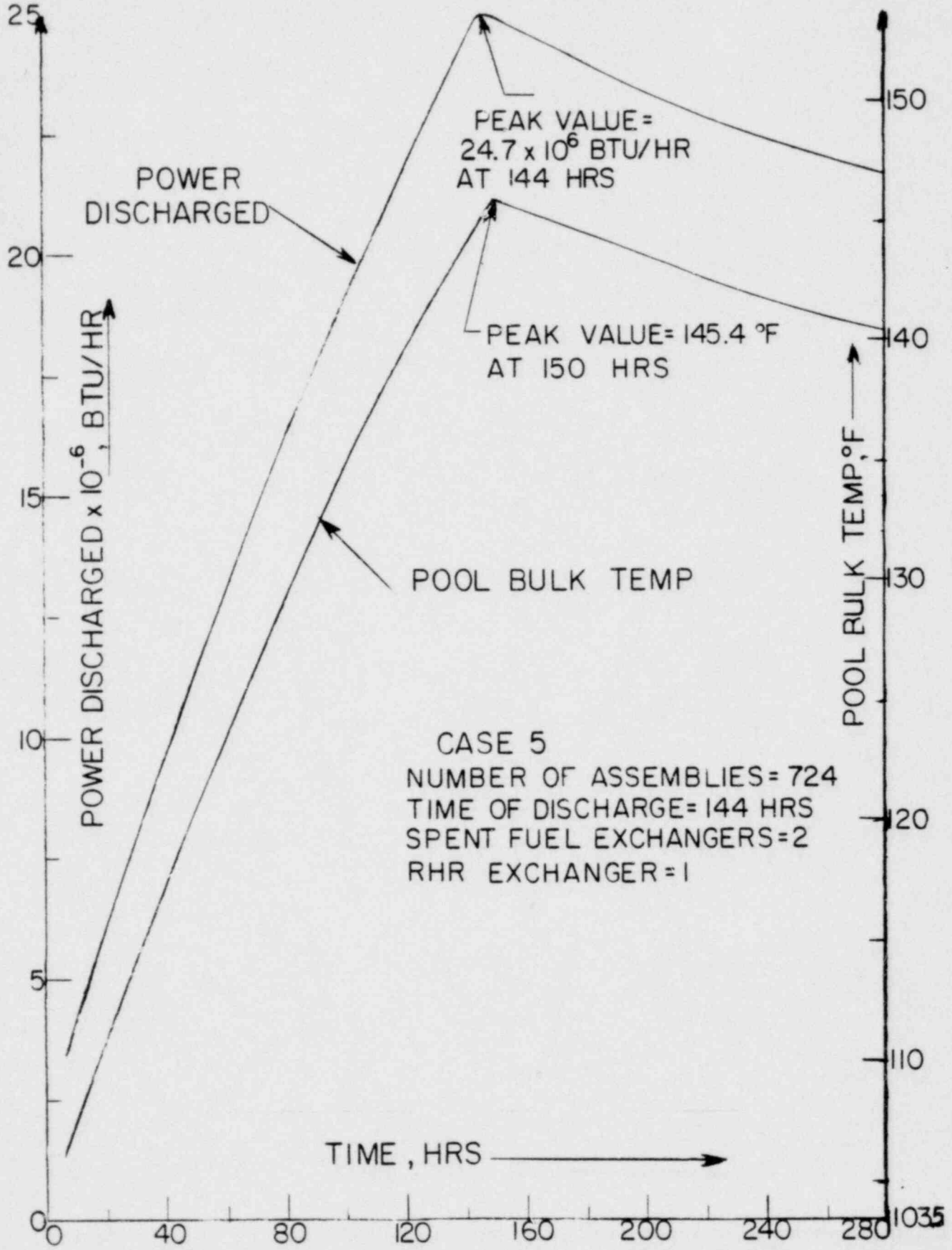

FIG 5.1.10
FULL CORE DISCHARGE,
724 ASSEMBLIES

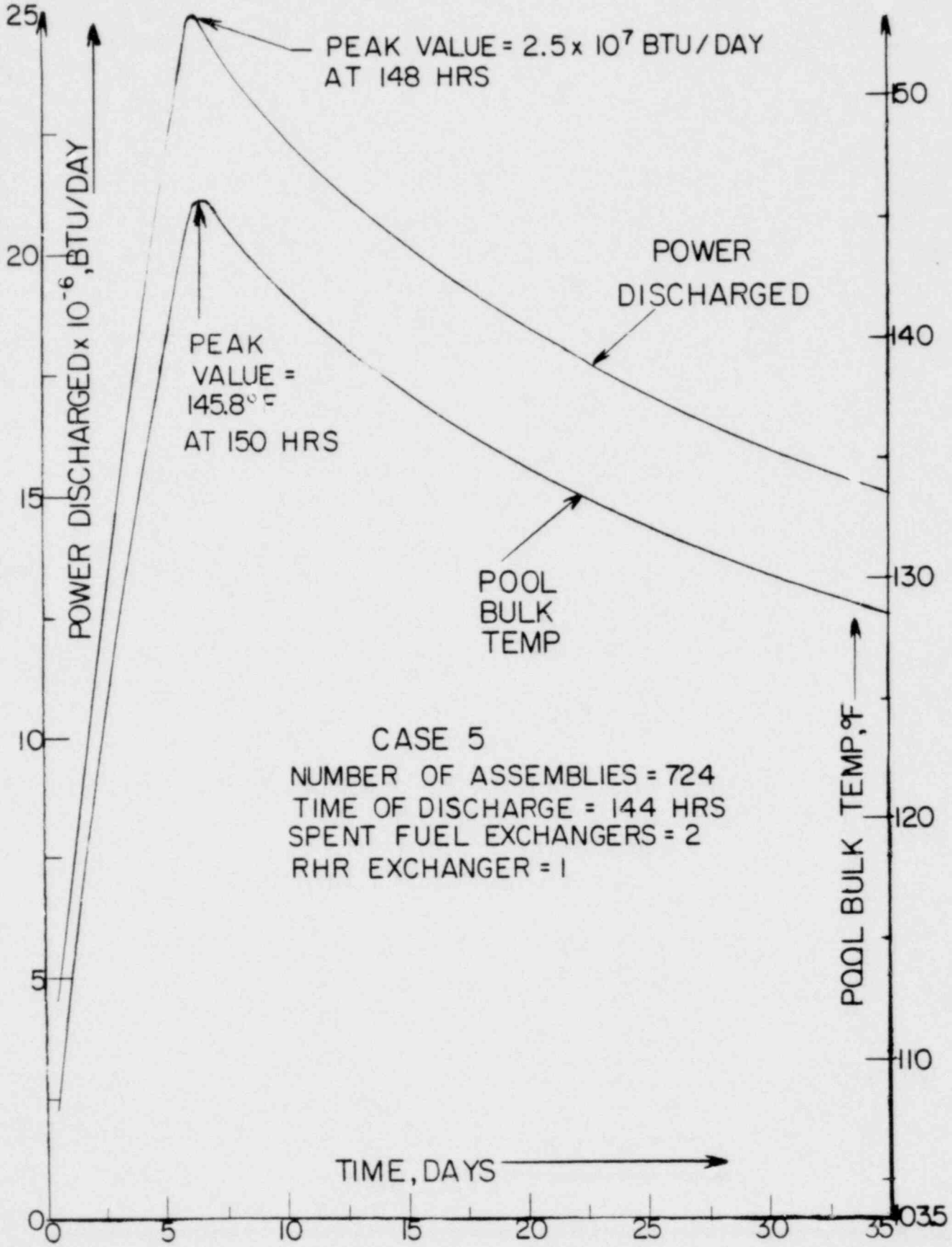

FIG 5.1.11
FULL CORE DISCHARGE,
724 ASSEMBLIES

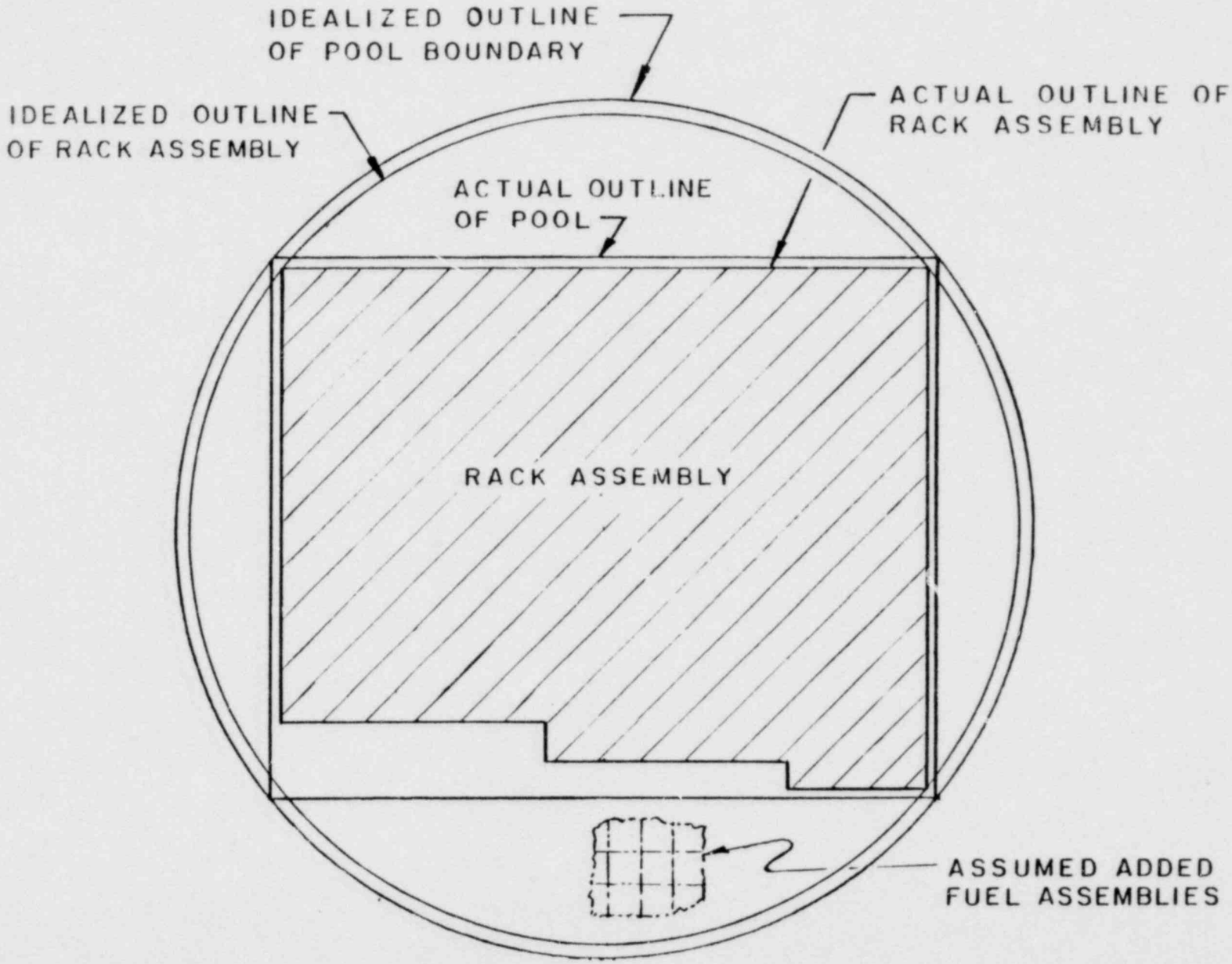

Figure 5.2.1 Rack Space Enveloping Cylinder
(Quad Cities Station Units 1 and 2)

8. RADIOLOGICAL CONSEQUENCES

8.1 Objectives and Assumptions

The radiological consequences of expanding the storage capacity of the spent fuel storage pool have been evaluated with the objective of determining if there is any significant additional radiological impact, onsite or offsite, relative to that of the currently authorized spent fuel storage pool. The principal factors considered in evaluating the additional radiological consequences were the following:

- o Operating experience and measurements.
- o Reduction in decay heat generating rate and fuel temperatures with time following removal from the reactor.
- o Age and nature of the additional fuel to be stored.
- o Analyses of radionuclide releases to the pool water from failed fuel.

In addition, the radiological impact to operating personnel has been evaluated to ensure that such exposure remains as low as is reasonably achievable.

Each spent fuel pool is currently authorized to store approximately two full cores. By comparison, each expanded spent fuel storage pool can accommodate more than five full core loads. The additional storage capacity will be used for aged fuel which has been out of the reactor 5 years or more. It is important to note that the difference between the radiological impact for the currently authorized storage pool capacity and the expanded storage pool capacity is attributable entirely to the presence of additional aged fuel in the expanded spent fuel storage pool. 2

The radiological consequences of storing the additional quantity of aged fuel have been evaluated. To ensure a conservative evaluation of the storage of failed fuel, it was assumed that the spent fuel storage pool is entirely filled with high-burnup spent fuel (28,500 Mwd/MtU burnup), ranging from newly removed fuel (1 core load of 724 fuel assemblies) to aged fuel with a cooling time of approximately 18 years. The maximum fission-product inventory in the stored fuel in each pool would result from an idealized fuel cycle in which approximately 181 spent fuel elements were removed from the core and placed in the pool annually. With this fuel cycle, the expanded storage pool capacity, when completely filled, would contain the following: 2

(1) For currently authorized storage capacity	724 newly removed assemblies (full core load) and 4 refueling discharges of 181 assemblies with storage periods of 1, 2, 3, and 4 years, respectively. 2
(2) Aged fuel in expanded storage capacity	13 refueling discharges of 181 assemblies with storage periods of 5 to 17 years and any remaining capacity (up to 170 assemblies), containing fuel stored for 18 years. 2

Reduced fuel burnup or increased cycle length would result in a lower fission-product inventory or longer storage (decay) periods. Thus, the assumed storage pool composition should result in a conservative estimate of any additional radiological impact due to the expanded storage capacity.

8.2 Operating Experience

8.2.1 General Industry Experience | 2

In a survey[1] of spent fuel storage pool experience, Johnson, at Battelle Pacific Northwest Laboratories, has shown that typical concentrations of radionuclides in spent fuel pool water range from 10^{-4} μCi/ml, or less, to 10^{-2} μCi/ml, with the higher value associated with refueling operations. Isotopic measurements of the nuclides confirm that a major fraction of the coolant activity results from activated corrosion products dislodged from fuel element surfaces during refueling operations or carried into the spent fuel pool water (with some fission-product radionuclides) by mixing the pool water with primary system water during refueling. These sources of storage pool radionuclides depend upon the frequency of refueling operations and are basically independent of the total number of fuel assemblies in storage.

Once fuel-handling operations are completed, the mixing of pool water with primary system water ceases and these sources of radionuclides decrease significantly; only dissolution of fission-products absorbed on the surface of fuel assemblies and low levels of erosion of corrosion-product (crud) deposits remain. With aged fuel (5 or more years storage), neither of these latter sources would be expected to contribute significantly to the concentrations of radionuclides in the storage pool.

In view of the above, it is concluded that the additional storage capacity of the expanded spent fuel pool will not measurably alter the currently approved radiological impact or impose any significant additional burden on the cleanup system as a result of corrosion-product radionuclides or fission-product carry-over from the primary system during refueling operations.

During storage, the level of gamma radiation from fission products in the fuel decreases naturally due to radioactive decay.

Because of this decay, the contribution of the aged fuel to the dose rate at the pool surface by direct radiation will be very small (< 5%) compared to that from the more-recently-discharged fuel. Thus, it is concluded that the occupational dose rate above the surface of the pool from direct radiation will be essentially the same as that for the currently authorized storage pool.

8.2.2 Related Plant Experience

8.2.2.1 Radionuclide Concentrations in Spent Fuel Pool Water

Measurements have been made of the principal radionuclide concentrations in both Quad Cities fuel storage pools during reactor operations. Table 8-1 summarizes these measurements. As shown in Table 8-1, the pool water radionuclide concentrations are not significantly affected by the number of fuel assemblies stored in the pool; over three (3) times as many fuel assemblies are stored in the QC unit 1 pool as in QC Unit 2 pool, but both pools have essentially the same Cs-134, Cs-137, and Co-60 radionuclide concentrations. This observation lends credibility to the expected low contribution from aged fuel in storage. 2

Similar measurements made at the Dresden Unit 2 pool (which is similar to the Quad Cities pools) indicate that the contribution, if any, from aged fuel will be very small or negligible in comparison to the higher activity levels (especially during refueling) of freshly discharged fuel. The Dresden measurements also show that the higher radionuclide concentrations which are measured during refueling operations rapidly (within 2 months) drop to near the pre-refueling levels even though the pool contains the freshly discharged fuel removed from the reactor.

With the expanded spent fuel storage capacity, the contribution from the aged fuel is correspondingly expected to be very low or negligible in comparison to that from recently discharged fuel or from primary system carry-over during refueling.

Table 8-1 Observed Radionuclide Concentrations In Spent Fuel Storage Pool Water

Date	Plant Status	Fuel Assemblies in Pool	μCi/cc Cs-134	Cs-137	Co-60
QC Unit 1					
4/27/81	Optg.	1139	2.6×10^{-4}	7.9×10^{-4}	7.1×10^{-5}
5/18/81	Optg.	1139	5.5×10^{-5}	1.8×10^{-4}	1.8×10^{-4}
5/24/81	Optg.	1139	4.5×10^{-5}	1.7×10^{-4}	1.9×10^{-5}
6/1/81	Optg.	1139	9.2×10^{-5}	3.0×13^{-4}	2.9×10^{-4}
QC Unit 2					
4/27/81	Optg	353	2.4×10^{-4}	7.8×10^{-4}	2.7×10^{-4}
5/18/81	Optg.	353	7.1×10^{-5}	2.2×10^{-4}	9.5×10^{-4}
5/24/81	Optg.	353	6.5×10^{-5}	2.6×10^{-4}	6.1×10^{-5}
6/1/81	Optg.	353	8.5×10^{-5}	2.8×10^{-4}	2.1×10^{-4}

8.2.2.2 Pool Cleanup System Operation

In the Quad Cities spent fuel storage pools, operation of the cleanup demineralizer system and frequency of resin replacement is determined primarily by requirements for water clarity rather than the loading of fission product radionuclides. The amount of suspended particulate material that must be removed to maintain the desired water clarity is determined by the frequency of refueling operations and is independent of the number of fuel assemblies stored. Thus, the expanded capacity of the Quad Cities storage pool will not significantly alter either the frequency of resin or filter media replacement above that currently experienced, or the personnel radiation exposures during maintenance operations. 2

8.2.2.3 Fuel Pool Radiation Levels

Measurements of the radiation levels above the spent fuel storage pool in both Quad Cities and the Dresden plants confirm that the dose rates are essentially independent of the number of fuel assemblies stored. On April 24, 1981, the measured dose rate above the Quad Cities unit 2 pool was 4 mr/hr with 353 assemblies in the pool and also 4 mr/hr above the Quad Cities Unit 1 pool with 1139 assemblies stored.

The average radiation dose above both Quad Cities pools during the period from January to April 1981 was 4-6 mr/hr. Somewhat higher radiation dose rates (up to 15 mr/hr) were observed above the Quad Cities pools during refueling operations, decreasing soon after completion of refueling to the 4-6 mr/hr range. Expanding the storage capacity of the spent fuel pools is thus not expected to significantly alter the radiation dose rates over the pools above that currently experienced.

In order to ascertain if there were crud depositions on the pool walls, measurements made above the center of the storage pool and at the pool edge were essentially the same, indicating that there are no significant crud depositions on the walls of the pool that might contribute to a higher dose rate at the pool edge. Visual observations also confirm the absence of any significant crud deposition on the pool walls. 2

Radiological surveys in the vicinity of the spent fuel storage pools indicate that the major sources of the observed dose rates are derived from miscellaneous pieces of equipment in the vicinity of the pool or utilized in fuel handling operations (e.g., sipping equipment, grapple attachments, vacuum hoses, etc.). None of these miscellaneous sources of radiation are affected by the number of fuel assemblies stored. Consequently, expanding the storage capacity of the Quad Cities spent fuel pool will not significantly alter the radiation dose to personnel occupying the fuel pool area.

8.2.2.4 Airborne Radionuclides

Because of radioactive decay, Kr-85 will be the only significant contributor to any potential increase in airborne radionuclide concentrations above that currently authorized. For the current Quad Cities storage pools, Kr-85 has not been detected at the reactor building vent (i.e., any Kr-85 present is less than the minimum detectable concentration of 6×10^{-6} μC/cc to 9×10^{-6} μC/cc). As discussed in Section 8.3.3 below, no significant increase in Kr-85 concentration from the aged fuel is expected. Consequently, expanding the spent fuel storage capacity will not impose any significant radiological burden from airborne radionuclides. 2

8.3 Consequences of Failed Fuel

Escape of fission-products from failed fuel stored in the spent fuel pool will contribute to the radionuclide concentrations in the pool water. However, calculations described below indicate that the radionuclide concentrations from failed fuel are considerably less than the concentrations of corrosion-product radionuclides and, therefore, the aged fuel in the expanded storage pool will not contribute significantly to the onsite or offsite radiological impact.

The decay heat generated in spent fuel rapidly decreases (by radioactive decay) following removal from the reactor and, in the aged fuel, will be very small (< 5% of that in freshly-removed fuel). Fuel temperatures and internal gas pressures will correspondingly decrease with time. Johnson[1] also cites evidence to confirm that UO_2 is inert to the relatively-cool water of spent fuel storage pools. Therefore, the release rate of fission-products from any defective rods among the aged fuel is expected to be negligibly small.

Release of fission-products from failed fuel probably results from water leaching or diffusion of material plated out or absorbed in

the fuel-clad gap of the fuel element during operation in the reactor. Once the material in the gap is depleted, further release will be very small. Most of the fission-products are absorbed (retained) in the fuel matrix and can escape only by diffusion through the UO_2. At the temperatures of the fuel in the spent fuel pool, the diffusion coefficient will be extremely small.[2]

In his survey, Johnson indicates that numerous fuel assemblies with one or more defects have been stored in several spent fuel pools without requiring special handling. Detailed analysis of the spent fuel pool water confirmed that fuel elements with defects do not continue to release significant quantities of radionuclides for long periods of time following removal from the reactor. Nevertheless, the calculations described here in Sections 8.3.1 and 8.3.2 were based on the very conservative assumption that the rate of fission-product release remains the same as the rate for newly discharged fuel.

Both Johnson, at Battelle, and Weeks,[3] at Brookhaven National Laboratory, have reviewed the corrosion properties of Zircaloy cladding and the integrity of spent fuel elements stored for long periods of time. They conclude that the corrosion of Zircaloy cladding in spent fuel pool water is negligibly small and that there is sufficient evidence of satisfactory fuel integrity to justify expanded storage. Consequently, there is not expected to be any significant deterioration of stored fuel that might lead to additional fuel failures in the expanded-capacity spent fuel storage pool.

8.3.1 Methods of Analysis

To assess the maximum potential radiological contribution from failed fuel, the inventory of fission-products in the spent fuel was calculated with the ORIGEN[4] code, conservatively assuming that all fuel was discharged from the core at 28,500 MWd/MtU burnup. Experimental values of escape rate coefficients in cool water shortly after

discharge, as derived by Westinghouse,[5] were used to calculate the fractional release of fission-products from failed fuel, and it was assumed that there were 1% fuel element failures. These escape rate coefficients (listed in Table 8-2) were assumed to be constant throughout the storage period, although it is known that fission-product release from failed fuel is strongly dependent upon the temperatures within the fuel pin. As natural radioactive decay occurs, decay heat generation in the fuel becomes less and, as a consequence, the fuel temperatures and internal gas pressures are reduced. Furthermore, the inventory of leachable fission-products becomes depleted and release from the bulk UO_2 by diffusion becomes extremely low.[3] | 2

Thus, within a few months after discharge, the fuel temperatures and effective leak rate coefficients decrease, and further leakage is reduced to relatively insignificant levels.[1]

The percentage of failed fuel that exists in the stored fuel, averaged over a large number of reactor cycles, is uncertain. Johnson[1] estimates that an average of 0.01% should be achievable. The NRC, in NUREG-0017, cites 0.12% as a representative value. Nevertheless, to establish a conservative upper limit, the calculations reported here were based on the assumptions that 1% of all stored fuel is failed and that constant leak rate coefficients, corresponding to those measured shortly after shutdown, apply over the storage periods. Concentrations of released fission-products were calculated from the dynamic balance between the source term (leakage from the assumed failed fuel) and the rate of removal by (1) radioactive decay and (2) the spent fuel pool cleanup system (using demineralizer cleanup efficiencies cited in NUREG-0017). This method of analysis is similar to that used in NUREG-0017.

Table 8-2 Escape Rate Coefficients into Cool Water for Spent Fuel in Storage Pool

Element	Escape Rate Coefficient (sec^{-1})
I	1.7×10^{-12}
Rb, Cs	3.0×10^{-12}
Mo	1.8×10^{-12}
Te	(0.9×10^{-12})*
Sr	8.5×10^{-15}
Ba	5.8×10^{-16}
Zr**	1.2×10^{-16}

*Escape rate coefficient for Te assumed to be in same ratio to Mo, as given in NUREG-0017.
**Assumed applicable to all other nuclides.

2

8.3.2 Fission-Product Radionuclide Concentrations

Based upon the method of analysis described above, the concentrations of fission-product radionuclides in the spent fuel pool were calculated at several times following unloading of a full core into the spent fuel pool, with the remainder of the pool assumed to be filled with older fuel. Results of these calculations are summarized in Table 8-3, assuming continual operation of the spent fuel pool water cleanup system. These calculated concentrations of fission-product radionuclides are directly proportional to the assumed 1% failed fuel and would be a factor of approximately 8 lower for the 0.12% failures estimated in NUREG-0017 as a typical weighted average value based on operating experience in a number of reactors. Of the fission-product radionuclides released, Cs-137 is the dominant activity from the aged fuel (calculated to be a maximum of 2.2×10^{-6} μCi/ml with 1% failed fuel). Low levels of I-131 (8×10^{-6} μCi/ml) and Mo-99 (3×10^{-6} μCi/ml) are calculated to be present as a result of leakage from a full core load of newly removed fuel (with 1% failures). However, in the aged fuel, these nuclides have decayed and there are no significant quantities of I-131 or Mo-99 remaining.

2

Table 8-3 Fission-Product Radionuclide Concentrations in Fully-Loaded Spent Fuel Storage Pool with 1% Failed Fuel 2

Time (days)	Concentration (μCi/ml) For Currently Approved Storage Capacity*	Incremental Additions due to Expanded Capacity*	Total
5	2.08×10^{-5}	2.62×10^{-6}	2.34×10^{-5}
10	1.43×10^{-5}	2.62×10^{-6}	1.69×10^{-5}
20	9.35×10^{-6}	2.61×10^{-6}	1.20×10^{-5}
30	7.54×10^{-6}	2.61×10^{-6}	1.02×10^{-5}
50	6.23×10^{-6}	2.60×10^{-6}	8.83×10^{-6}
75	5.67×10^{-6}	2.59×10^{-6}	8.26×10^{-6}
100	5.37×10^{-6}	2.57×10^{-6}	7.94×10^{-6}

*See Section 8.1 for a description of composition.

Even with 1% failed fuel, the radionuclide concentrations in the spent fuel pool water are dominated by those from corrosion products and carry-over from the primary coolant system during refueling. Furthermore, since the release rate from the aged fuel will be considerably smaller than that indicated in Table 8-3 (due to the lower temperatures and release rates in the fuel elements), the actual contribution from the aged fuel will be negligibly small in practice. It is also expected that the percentage of failed fuel, averaged over the reactor lifetime, will be considerably less than 1%. Thus, it is concluded that the expanded-capacity spent fuel storage pool will not increase the radionuclide concentrations in the pool water significantly above those for the currently approved spent fuel storage pool. Consequently, expanding the storage capacity of the spent fuel pool will neither alter the onsite or offsite radiological impact nor significantly increase the burden on the spent fuel pool cleanup system, as a result of failed fuel. 2

8.3.3 Gaseous Releases from Failed Fuel

Because of the half-lives of the noble-gas radionuclides, only the release of Kr-85 ($T_{\frac{1}{2}}$ of 10.76 years) has the potential of

increasing the radiological impact to the reactor building atmosphere as a result of expanding the capacity of the spent fuel storage pool. (Short-lived noble-gas radionuclides and other volatile fission-products, such as iodine, are not present in the aged fuel.) Johnson[1] concludes that the radioactive fission gases will have been largely expelled from defective fuel rods during reactor operation and, therefore, are not available for release during fuel storage. This is expected, since the noble gases are chemically inert and there are no plate-out or hold-up mechanisms in the fuel-clad gap of the fuel element. Measurements above the Quad Cities storage pools failed to detect any Kr-85 above the minimum detection level. | 2 | 2

The small amount of chemically inert Kr-85 that might be absorbed on the surface of a fuel assembly and released slowly during storage, is believed to be insignificant, particularly in the aged fuel. Since UO_2 is chemically inert to cool water, diffusion of Kr-85 entrapped within the UO_2 fuel matrix would be the remaining source for Kr-85 release. Based on the method outlined in the proposed ANS 5.4 standard[2] on fission gas release, the diffusion coefficient in the aged fuel at spent fuel pool temperatures will be negligibly small (of the order of 10^{-40}). Consequently, diffusion release of Kr-85 from aged fuel will be negligible in accord with Johnson's findings.[1]

It is concluded that the incremental radiological impact from the release of Kr-85 with the expanded-capacity spent fuel storage pool will be negligibly small.

8.4 Exposure for the Installation of New Racks

The existing spent fuel racks will be removed, and the new racks will be installed in a manner which will maintain occupational exposure to levels as low as reasonably achievable (ALARA). The following

methods for the disposal of the existing racks are currently under review by CECo:

- o Crating and shipment of the racks in "as-is" condition.
- o Decontamination and shipment.
- o Dismantle and volume reduction, with or without prior decontamination, and shipment of waste.

The final decision concerning the disposal of the existing spent fuel racks will be based on project needs and experience gained from rack disposal at Dresden.

8.5 Conclusions

Based on operating experience and the analysis of potential releases, it is concluded that expanding the capacity of the spent fuel storage pool will not significantly increase the onsite or offsite radiological impact above that of the currently authorized storage capacity. Similarly, the expanded storage capacity will not impose any significant additional burden on the spent fuel pool cleanup system, and no modifications to the current radiation protection program are needed. 2

REFERENCES TO SECTION 8

1. A. B. Johnson, Jr., "Behavior of Spent Nuclear Fuel in Water Pool Storage," BNWL-2256, September 1977.

2. ANS 5.4 Proposed Standard, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel."

3. J. R. Weeks, "Corrosion of Materials in Spent Fuel Storage Pools," BNL-NUREG-2021 (Informal Report), July 1977.

4. M. J. Bell, "ORIGEN-The ORNL Isotope Generation and Depletion Code," ORNL-4628, May 1973.

5. J. M. Wright, "Expected Air and Water Activities in the Fuel Storage Canal," WAPD-PWR-CP-1723 (with addendum), undated.