CP&L

Carolina Power & Light Company

Regulatory File Cy

September 5, 1975

File: NG-3514 (R)

Serial: NG-75-1392

DOCKETED USNRC SEP 8 1975 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

RE: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
LICENSE NO. DPR-23
SPENT FUEL STORAGE CAPACITY EXPANSION

Dear Mr. Rusche:

Carolina Power & Light Company is planning to expand the storage capacity of the spent fuel storage pool at the H. B. Robinson Unit 2. The Company has determined that the expansion of the storage capacity represents an unreviewed safety matter as specified in Section 50.59(c) of 10CFR50. Therefore in accordance with Section 50.90 of 10CFR50, Carolina Power & Light Company requests an amendment to Facility Operating License DPR-23 to permit installation and use of the new spent fuel racks.

Enclosed are three signed originals of the license amendment application and forty (40) copies of a report titled, "H. B. Robinson Unit 2 Spent Fuel Storage Expansion," which describes the modifications and the supporting analyses for your review. The Plant Nuclear Safety Committee has reviewed the report and approved it. The Company Nuclear Safety Committee has also reviewed the report and concurs with the report.

Approval of the new fuel racks and their installation must be accomplished by December 1, 1975, to insure fabrication and installation prior to refueling in Fall, 1976. Therefore, a prompt review of the report and approval of the modifications are requested.

Yours very truly,

E. E. Utley
Vice President
Bulk Power Supply

RECEIVED SEP 8 1975 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

RLM/kr
Enclosure
cc: Messrs. N. B. Bessac, P. W. Howe, W. B. Kincaid, J. B. McGirt, R. E. Jones, S. McManus, W. W. Reynolds, L. E. Smith, D. B. Waters

Sworn to and subscribed before me this 5th day of September 1975.

Nancy A. Stephens (Yancey)
Notary Public

My Commission expires: June 29, 1976

NANCY A. STEPHENS NOTARY PUBLIC WAKE COUNTY, N.C.

336 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

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H. B. ROBINSON UNIT 2
SPENT FUEL STORAGE EXPANSION

H. B. ROBINSON UNIT 2

SPENT FUEL STORAGE EXPANSION

Table of Contents

1.0 INTRODUCTION

1.1 HISTORY AND NEED FOR ADDITIONAL SPENT FUEL RACKS

The H. B. Robinson spent fuel pool was built to accommodate 240 fuel assemblies or about 1-1/2 cores for equilibrium cycle operation assuming 1/3 core is discharged and shipped for reprocessing each year. The original capacity would permit the core to be unloaded with a recently discharged batch still present in the pool.

As a consequence of delays being experienced by CP&L's spent fuel reprocessor, Allied General Nuclear Services (AGNS), and the early discharge of a low-pressurized batch, there are presently 156 assemblies stored in the pool (one assembly was shipped to Aerojet Nuclear Company for use in an NRC safety research program). The existing capacity will permit only one additional batch to be discharged. This is currently scheduled for November, 1975.

Since substantial uncertainties exist in the present schedule for reprocessor availability, an additional 36 storage locations will be installed in the available space along the west end of the pool. This will provide sufficient spent fuel storage capacity for the plant to operate until the Fall of 1977. By that time, spent fuel presently in the pool can be shipped to AGNS facilities, or an alternative will be developed.

2.1 DESCRIPTION AND LOCATION OF RACKS

The 36 additional spent fuel storage locations (cells) will be provided in 2 rack modules, each holding 18 fuel assemblies. The design of these cells will be similar in concept to the existing racks; i.e., open lattice type construction will be used. However, the center-to-center spacing (pitch) of the cells in the new racks will be 15.5 inches instead of the 21-inch pitch used in the existing racks. The new racks will be installed in a presently-vacant space along the west wall of the spent fuel pit, as shown in Figure 2.1-1.

The new racks will be constructed entirely of Type 304 stainless steel compatible with the existing racks and pit liner which are made of the same material. The racks are designated as seismic Category I, meeting the requirements of NRC Regulatory Guide 1.29. The design will comply with the American Institute of Steel Construction (AISC) "Specification for the Design, Fabrication & Erection of Structural Steel for Buildings." Additional structural design criteria are provided in Section 3.2.2. Welding will be in accordance with Section IX of the ASME Boiler and Pressure Vessel Code. Rack design and fabrication will be performed using CP&L's QA program.

Fuel assemblies will sit in the racks in a vertical square array. The rack design precludes inserting a fuel assembly in other than a prescribed location, assuring that the designed pitch of 15.5 inches is maintained and thus keeping the fuel in a subcritical array. An entry guide at the top of each cell is provided to guide the fuel assembly into that cell without damage. The cells are constructed of structural members which support the fuel assembly at the top and bottom nozzles only, and which permit the flow of coolant around the entire fuel assembly. The inside surfaces of each cell are free of rough edges which might damage fuel. The racks will sit on the floor of the spent fuel pit on supports which will adjusted to assure a level orientation. Seismic support is provided by lateral supports which transmit horizontal seismic loads into the spent fuel pit walls. There will be no attachments to the liner of the spent fuel pit.

2.2 Installation Procedures and Schedule

2.2.1 Controls for CP&L activities are outlined in the CORPORATE QUALITY ASSURANCE PROGRAM, PART 2. PART 2, Section 3, Operation and Maintenance Modification Control, sets forth the requirements for changes such as the addition of spent fuel storage racks to H. B. Robinson Unit 2. It includes requirements for the accomplishment of plant modifications, and instructions for the preparation and issuance of modification procedures. Such procedures are developed for each plant modification to assure that all necessary activities are carried out in a planned, controlled, and orderly manner. As appropriate to the circumstances, this procedure will:

a. Define necessary prerequisites.
b. Present a chronological sequence of events appropriate to the circumstances.
c. Specify all appropriate special processes.
d. Reference applicable drawings.
e. Specify the use of special equipment and tools, including calibration requirements as appropriate.
f. Include applicable inspection and testing acceptance criteria.
g. List the codes, standards, and regulations which apply to the modification.

The chronological sequence section of the procedure will specify a detailed step-by-step procedure for performing this change. An outline of these steps is as follows:

a. Orient, train, and qualify personnel for installation of the racks.
b. Fabricate or prepare a lay-down area in the plant for temporary storage of the racks.

c. Fabricate a lay-down area in the Fuel Handling Building.

d. Erect temporary lifting mechanism with special harness and tools for lowering the racks into the pool.

e. Transfer racks from carrier to outside lay-down area.

f. Remove and secure portable wall and roof panels in the Spent Fuel Building.

g. Transfer the racks to the lay-down area inside the Spent Fuel Building.

h. Inspect the racks for damage.

i. Transfer the racks to the temporary lifting mechanism.

j. Lower the racks into place in the spent fuel pool.

k. Inspect and verify that racks are ready for operational test.

l. Test racks.

m. Clear the racks for operation.

The procedure will provide for a redundant lifting arrangement for attaching the racks to the crane hook. This arrangement will preclude the unlikely event of dropping a rack during installation. In addition, the lifting mechanism will be restrained to prevent the possibility of falling into the pool.

2.2.2 The present schedule for installation is as follows:

Receive racks at plant	May 1976
Complete installation and testing	July 1976
Refueling Outage (earliest date)	September 1976

3.0 DESIGN CRITERIA AND SAFETY ANALYSES

The modification described earlier in this report was selected because it provides the additional fuel storage required for the 1976 refueling, while maintaining a conservative design. However, since the 15.5 inch pitch represents a 5.5 inch reduction in pitch from the existing racks, and since the modification results in somewhat higher heat loads being imposed on the spent fuel pool, analyses have been performed to verify that no aspects of the modification represent an undue risk to public health and safety. The following sections describe these analyses.

3.1 CRITICALITY ANALYSIS

3.1.1 Assumptions

The following assumptions were used in the criticality study:

1. The fuel assembly design was a 15 x 15 array of 95% dense UO_2 rods.
2. All fuel rods were unirradiated with an enrichment of 3.20 w/o U-235. Since the maximum enrichment used thus far at H. B. Robinson has been 3.12 w/o, the selection of 3.20 w/o is conservative. Furthermore, projections indicate that future reload enrichments will be the range of 2.9 to 3.0 w/o.
3. The reference moderator and fuel temperature for the criticality calculation is 125°F. Calculated values of K_{eff} versus temperature show that K_{eff} is the largest at this temperature. (See Fig. 3.1-2)
4. To maximize the calculated reactivity, zero current boundary conditions were used to simulate an infinite array in the X-Y direction and axial leakage was not allowed.
5. The only structural materials used in the analysis were the fuel assembly grids and the stainless steel corner angles constituting each cell.
6. No soluble boron, burnable poison rods, or other fixed poisons were considered in the calculations.

3.1.2 Nuclear Methods

Four group cross sections were generated using the XPOSE computer code. The thermal energy cutoff was 1.85 ev. XPOSE is an Exxon Nuclear modified version of LEOPARD and is described in XN-CC-21, "XPOSE - The Exxon Nuclear Revised LEOPARD." These cross sections were used in a two-dimensional neutron diffusion program PDQ-7 with the lattice geometry shown in Figure 3.1-1. Two mesh intervals per fuel rod were used in the fuel region for the PDQ calculation. Variable mesh spacing was used in the moderator region depending

upon center-to-center spacing. The stainless steel structural supports were represented discretely.

3.1.3 Results

The variation of K_{eff} at the 15.5 inch spacing with temperature is shown in Figure 3.1-2. In these calculations, both fuel and moderator temperatures were varied. Thus, the slope of the curve is the isothermal temperature coefficient. The very small positive coefficient from 50°F to 125°F is due to the over-moderated lattice condition. Beyond 125°F, the lattice exhibits a negative coefficient. The positive isothermal temperature coefficient (maximum ΔK increase of 0.0015) is not considered significant when compared to the subcritical condition of the array.

Figure 3.1-3 shows the calculated K_{eff} as a function of center-to-center spacing at a temperature of 125°F. This temperature corresponds to the highest reactivity condition at a spacing of 15.5 inches. For the design value of 15.5 inch spacing, the calculated K_{eff} is 0.896. A worst case K_{eff} can be determined from the effects of (1) dimensional tolerance on the nominal 15.5 inch spacing and (2) calculational uncertainties. This worst case K_{eff} can then be compared to the design limit K_{eff} of 0.95 as given in ANSI N18.2 .

Based upon dimensional tolerances, the minimum center-to-center spacing between adjacent racks is 14.86 inches. Figure 3.1-3 shows K_{eff} is 0.910 at this spacing. Calculational uncertainties can be attributed to both the nuclear methods and the absorbing characteristics of the structural materials. An estimate of the uncertainty in nuclear methods can be made with reference to benchmark calculations on critical experiments. The XPOSE code used in this study has been compared against critical experiments in XN-75-27, "Exxon Nuclear Neutronic Design Methods for Pressurized Water Reactors." In thirty-four critical experiments described in this report, the standard deviation in ΔK is 0.3%.

At a 95% confidence level, the nuclear methods uncertainty is 2 x 0.3% = 0.6%. Since the total worth of the structural stainless steel is only 0.015 ΔK, a 25% error of 0.004 ΔK is the maximum uncertainty that can be expected. The worst case K_{eff} then becomes the following:

K_{eff} at 14.86 inch spacing	0.910
Nuclear methods uncertainty	+0.006
Structural absorbing uncertainty	+0.004
Worst case K_{eff}	0.920
Design limit	0.950
Margin	0.030

The results show that substantial margin exists between the worst case K_{eff} and the 0.95 design limit from ANSI N18.2.

3.1.4 Conclusions

The criticality study has shown that the fuel storage racks can be designed for a 15.5 inch spacing with assurance that a K_{eff} less than 0.95 is maintained. The calculated margin of 0.030 ΔK is more than sufficient considering the conservative assumptions used in this analysis. Thus, the design complies with General Design Criteria 62 of Appendix A to 10CFR50. (At the time the FSAR was submitted, the applicable Criterion Number was 66.)

3.2 STRUCTURAL AND SEISMIC DESIGN CRITERIA

3.2.1 Spent Fuel Pit

The Spent Fuel Pit was designed as a seismic Category I structure. Design criteria for Category I structures are given in Appendix 5A of the H. B. Robinson FSAR. The original design of the Spent Fuel Pit considered loads due to the water, spent fuel assemblies, spent fuel racks, and seismic activity. The analysis was performed using the design criteria contained in ACI-318-63, "Building Code Requirements for Reinforced Concrete."

Originally, the Spent Fuel Pit was designed for spent fuel racks to occupy the entire floor of the pit. However, only 240 cells were provided, leaving empty the rectangular corridor along the west wall. Thus, the floor of that corridor was designed to support spent fuel racks with 21" center-to-center cell spacing. The load-carrying capability of that floor with the additional 36 cells on 15.5" centers has been evaluated by the original design organization, Ebasco Services. The results of the evaluation show that the existing Fuel Handling Building is structurally adequate to withstand the load imposed by the additional 36 cells and associated fuel, and that design margins have been decreased only slightly.

3.2.2 Spent Fuel Storage Racks

The spent fuel storage racks to be installed are designed to remain in position during all operating modes and to transmit any loads to the structure of the Fuel Handling Building. Structural design of the racks is such that deformations are limited to preclude any possibility of criticality for all anticipated loadings including thermal expansion, dead loads, seismic forces, and dropped fuel assemblies. Specifically:

a. The storage racks are designed for thermal expansion loads imposed by a pool water temperature of 180°F.

b. Fuel assembly dead loads are based on a dry weight of 1605 pounds for each fuel assembly (which includes the maximum insert weight).

c. The storage racks and supports are designated as Seismic Category I and are designed to withstand the effects of the safe shutdown earthquake while loaded with fuel and remain functional, consistent with the requirements of NRC Regulatory Guide 1.29.

d. The racks are designed to preclude the centerline-to-centerline distance between adjacent storage spaces from changing due to the impact from fuel assembly dropped from a height 6" above the top of the racks.

The allowable stress criteria defined in Part 1 of the AISC "Specification for the Design Fabrication, and Erection of Structural Steel for Buildings," (1969) was used. However, the one-third increase in allowable stresses due to seismic loading allowed by Section 1.5.6 of that specification was not used.

3.3 SPENT FUEL POOL COOLING CAPABILITY

3.3.1 Design Requirements

The following requirements are applicable to the evaluation of the existing Spent Fuel Pool (SFP) Cooling System with the increased heat load of additional stored fuel assemblies.

a. The SFP cooling loop shall be capable of removing the heat generated by stored spent fuel elements from the spent fuel pool following either refueling or core unload situations.

b. The SFP, Component Cooling Water (CCW), and Service Water (SW) cooling loops shall maintain the SFP water at a temperature equal to or less than 180°F.

c. The heat transfer capability of the SFP Cooling System shall be based upon a maximum service water inlet temperature (medium to which the heat is ultimately transferred) of 95°F.

d. The SFP Cooling System shall have a means by which alternate cooling can be provided within the time period of thermal inertia (time for pool water temperature to rise to 180°F) of the pool.

These requirements satisfy applicable requirements of Criterion 61 and 63 of Appendix A to 10CFR50. (At the time the FSAR was submitted, the applicable Criterion number was 67.)

3.3.2 Component Cooling and Service Water Cooling Systems

The SFP heat exchanger (HX) uses component cooling water (CCW) to cool the SFP water. In turn, the CCW HX is cooled by the Service Water System. The CCW system and SW system are discussed in detail in Sections 9.3 and 9.6.2 of the H. B. Robinson Unit No. 2 FSAR.

The temperature of the service water going into the CCW HX is a controlling factor in establishing the heat transfer capability of the SFP cooling system. Table 3.3-1 presents pertinent component cooling heat exchanger design data, and Figure 3.3-1 is a simplified flow diagram of the 3 systems. A maximum service water inlet temperature to the CCW heat exchangers of 95°F was used in these analyses consistent with the original design bases given in the FSAR.

3.3.3 SFP Cooling System

The SFP HX, pumps, and piping constitute the mechanical means for removing heat from the SFP. Table 3.3-2 presents pertinent SFP heat exchanger design data and Figure 3.3-2 depicts a flow diagram of the major components of the system. In addition to this cooling, a certain amount of heat loss is obtained by evaporation from the pool surface into the air above the SFP, but this is not taken into consideration in the following analysis in order to provide an additional conservatism. A detailed description of the SFP cooling system is contained in Section 9.3 of the Robinson FSAR.

3.3.4 Heat Load to SFP Cooling System

The proposed addition of SFP storage racks will increase the storage capacity of the pool and, therefore, the total heat load which the SFP Cooling System must dissipate. The total heat load which the SFP Cooling System will be required to remove was calculated for two different cases. These cases are as follows:

Case 1 - Refueling Case. This case models the situation which is expected to exist during the Fall 1976 refueling outage.

a. The pool already contains 209 fuel assemblies from three previous refueling outages. Of these 209 assemblies, 53 have cooled approximately 3.5 years, 104 have cooled for approximately 2.5 years, and the remaining 52 cooled for approximately 1 year.

b. During this outage, 52 additional assemblies will be moved into the pool.

Case 2 - Equilibrium Cycle, Core Unload Case. This case models the worst case which might occur at some point in the future.

a. The pool already contains 104 fuel assemblies from two previous refueling outages. Of these 104 assemblies, 52 have cooled for approximately 2 years, and 52 have cooled for approximately 1 year. Each of the 104 assemblies was irradiated through three equilibrium cycles.

b. During this outage, the entire 157 assemblies of the core are transferred to the spent fuel pool; 53 of these have been irradiated through three equilibrium cycles, 52 through two equilibrium cycles, and 52 through one equilibrium cycle.

The decay heat rate was calculated as a fraction of operating power for all 261 assemblies utilizing the model presented in the October 1973 draft of ANS 5.1, Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors. To assure conservatism, these decay heat rates were multiplied by 1.10 to provide heat loads on the SFP Cooling System.

The total heat load in the spent fuel pit was calculated by first multiplying the power fractions (as calculated using ANS 5.1) times the average assembly power level at full power, and then summing the contribution from each assembly in the pool. An average assembly power level at full power of 14.01 MWt was used for Case 1. For Case 2 an average assembly power level of 14.65 MWt (2300 MWT reactor power which assumes the plant is uprated by 100 MWt) was used.

The Technical Specifications require a waiting period of 100 hours between reactor shutdown and the commencement of fuel removal from the core. In addition to this, approximately twenty minutes is required to transfer each assembly from the reactor to the spent fuel pool. During the 100-hour cooldown period and the period of transfer, the fuel in the reactor vessel is cooled by the Residual Heat Removal (RHR) system. Therefore, the entire heat load is not carried by the SFP Cooling System for a period of 118 hours after shutdown in Case 1, or for a period of 154 hours for Case 2.

The results of this calculation are shown in Figure 3.3-3 for Case 1 and Figure 3.3-4 for Case 2. These curves were generated by computing the spent fuel pool heat loads, in the manner previously discussed, at 7-day intervals beginning upon completion of fuel transfer and continuing for one year.

3.3.5 SFP Temperatures

Heat balances, based on heat loads shown in Table 3.3-3 and heat exchanger design data shown in Tables 3.3-1 and 3.3-2, were performed on the entire cooling system for both the refueling case and the core unload case. The results of the two heat balances are shown in Table 3.3-4. As shown in the table, the maximum SFP water temperature will be about 125°F for the normal refueling situation and about 162°F for the core unload situation. Because of the conservatisms in the analysis (e.g., ANS 5.1) upper limit margins on the decay heat, minimum achievable fuel movement times, and maximum service water temperature) it is expected that the SFP water temperatures will never reach these values. For the refueling case, the 125°F water temperature is not a significant change from the 120°F for which the SFP Cooling System was designed (shown on FSAR Table 9.3-3). For the core unload case, it will be desirable to maintain the SFP below 150°F due to thermal inertia considerations, as shown subsequently.

There will be no adverse effect on the SFP cooling loop components or any other affected components in the CCW and SW cooling systems as the design temperature for these items is 200°F or higher.

3.3.6 Thermal Inertia

The thermal inertia of the pool is defined as the time it would take after a loss of cooling for the pool water to reach a temperature of 180°F. The 180°F limitation is consistent with the FSAR. It is assumed that for either case being considered (refueling or core unload), cooling is lost when the last fuel assembly is inserted into the SFP. The heat inputs to the water are then at their maximum and are obtained from Table 3.3-4.

The parameters used for calculation and the resulting thermal inertia in hours are presented in Table 3.3-5. The time intervals presented are conservatively short as credit has not been taken for cooling by evaporation from the pool surface or the thermal inertia of the steel rack components or pool walls. Neither was credit taken for the more significant conservatisms introduced by ANS 5.1 in the heat load calculations.

Alternate Cooling Capabilities

There are two alternative means of providing cooling for the spent fuel pool in the event of failure of SFP Cooling System components:

a. A redundant SFP pump is being permanently installed in parallel with the existing SFP pump. In the event of the failure of the operating SFP pump, the redundant pump will be started.

b. In case of loss of component cooling water, water from the fire protection system can be connected to the shell side of the SFP heat exchanger.

Considering the thermal inertia of just over 1.5 hours that we have projected for worst case, it will be desirable to prevent the temperature of the SFP from reaching 162°F. Therefore, the rate of fuel movement into the Spent Fuel Pit will be regulated to maintain the SFP temperature at or below 150°F.

For both cases, there is sufficient thermal inertia to allow either of the alternative cooling means to be implemented.

3.4 RADIOLOGICAL CONSEQUENCES

The principal source of radiation exposure levels at the surface of the spent fuel pool is the concentration of radionuclides within the pool water during approximately the first 100 days following a refueling. These radionuclides are removed from the water by the spent fuel pool demineralizer during and after refueling. After the first 100 days, the demineralizer is required only to maintain water clarity. After this point exposure levels at the surface of the pool become insignificant. Because of the small radionuclide contribution of older spent fuel, the increase in fuel storage resulting from this modification should not significantly affect the pool surface radiation levels.

In the existing spent fuel storage racks, the sides of fuel assemblies stored closest to the wall are approximately 27" from the stainless steel plate liner. The new racks will be approximately 5 1/2" from the wall at the closest point. Radiation levels outside the spent fuel pit walls will remain lower than natural background since the 6' thick concrete walls provide most of the shielding.

The additional fuel storage resulting from this modification should have no effect on airborne radioactivity and the consequences of fuel handling accidents previously analyzed for this plant. This is the case since this modification does not change any of the assumptions or conditions used in those analyses.

3.5 SPENT FUEL CASK DROP

Provisions have been made to eliminate the spent fuel cask drop as a credible accident. Redundancy has been incorporated in the design of the spent fuel cask lifting lugs, lifting rig, and the 125-ton spent fuel cask handling crane to eliminate any risk to public health and safety. A detailed discussion of the safety features of the cask and handling components was supplied to the NRC in our letter of October 17, 1974. In addition, the spent fuel cask is never transported over stored fuel.

The addition of 36 more spent fuel assemblies to the spent fuel pool will not affect the spent fuel cask drop accident analysis.

TABLE 3.3-1

COMPONENT COOLING HEAT EXCHANGER DESIGN DATA

Parameter	Shell (Component Cooling Water)	Tube (Service Water)
Flow Rate, lbs/hr	4.46×10^6	4.96×10^6
Inlet Temperature, °F	115	95
Outlet Temperature, °F	108	101
Design Temperature, °F	200	200
Surface Area, ft^2	---	7760
Heat Exchanged, BTU/hr	29.4×10^6	---

TABLE 3.3-2

SPENT FUEL POOL HEAT EXCHANGER DESIGN DATA

Parameter	Shell (CCW)	Tube (SFP Water)
Flow Rate, lbs/hr	1.4×10^6	1.1×10^6
Inlet Temperature, °F	100	120
Outlet Temperature, °F	106	113
Design Temperature, °F	200	200
Surface Area, ft^2	---	2000
Heat Exchanged, BTU/hr	7.96×10^6	---

TABLE 3.3-3
HEAT LOAD DISTRIBUTION
(BTU/Hr)

Cooling System	Normal Refueling[1]	Core Unload[2]
SFPCW	9.50×10^6	24.5×10^6
RHR	17.23×10^6	0
CCW	26.73×10^6	24.5×10^6
SW	26.73×10^6	24.5×10^6

Notes: [1]During refueling, nonessential heat loads may be isolated from the CCW system.

[2]After core unload, the entire core is in the spent fuel pit, and consequently, RHR heat load is zero. Nonessential heat loads may be isolated from the CCW system.

TABLE 3.3-4
HEAT BALANCE & RESULTS

Component(s) in Active Service	Normal Refueling	Core Unload
CCWHX Heat Removal Rate (ea. of 2 HXs)	13.36×10^6 Btu/hr	12.25×10^6 Btu/hr
RHRHX Heat Removal Rate	17.23×10^6 Btu/hr	0
SFPHX Heat Removal Rate	9.5×10^6 Btu/hr	24.5×10^6 Btu/hr
CCWHX SW Inlet Temperature	95°F	95°F
CCWHX SW Outlet Temperature	98°F	97°F
CCWHX CCW Inlet Temperature	104°F	103°F
CCWHX CCW Outlet Temperature	100°F	101°F
SFPHX CCW Inlet Temperature	101°F	101°F
SFPHX CCW Outlet Temperature	108°F	118°F
SFPHX SFPCW Inlet Temperature	125°F	162°F
SFPHX SFPCW Outlet Temperature	116°F	139°F

Notes:
1. Heat loads given for the refueling case are for the point in time 118 hours after reactor shutdown, i.e., the time of maximum spent fuel pool water temperature for that case.
2. Heat loads given for the core unload case are for the point in time 154 hours after reactor shutdown, i.e., the time of maximum spent fuel pool water temperature for that case.

TABLE 3.3-5

THERMAL INERTIA OF SFP

I. Parameters Used

a) Volume of water, ft^3 — 36,612[1]
b) Specific heat of water, Btu/lb-°F — 1.0
c) Heat input, Btu/hr x 10^6

1. Refueling Case 9.5
2. Core Unload 24.5

II. Thermal Inertia, hours

a) Refueling, 125°F to 180°F: — 13.41
b) Core Unload, 162°F to 180°F — 1.58

Notes: [1]This volume of water is the actual pool volume corrected for the presence of existing and new spent fuel storage racks and fuel.

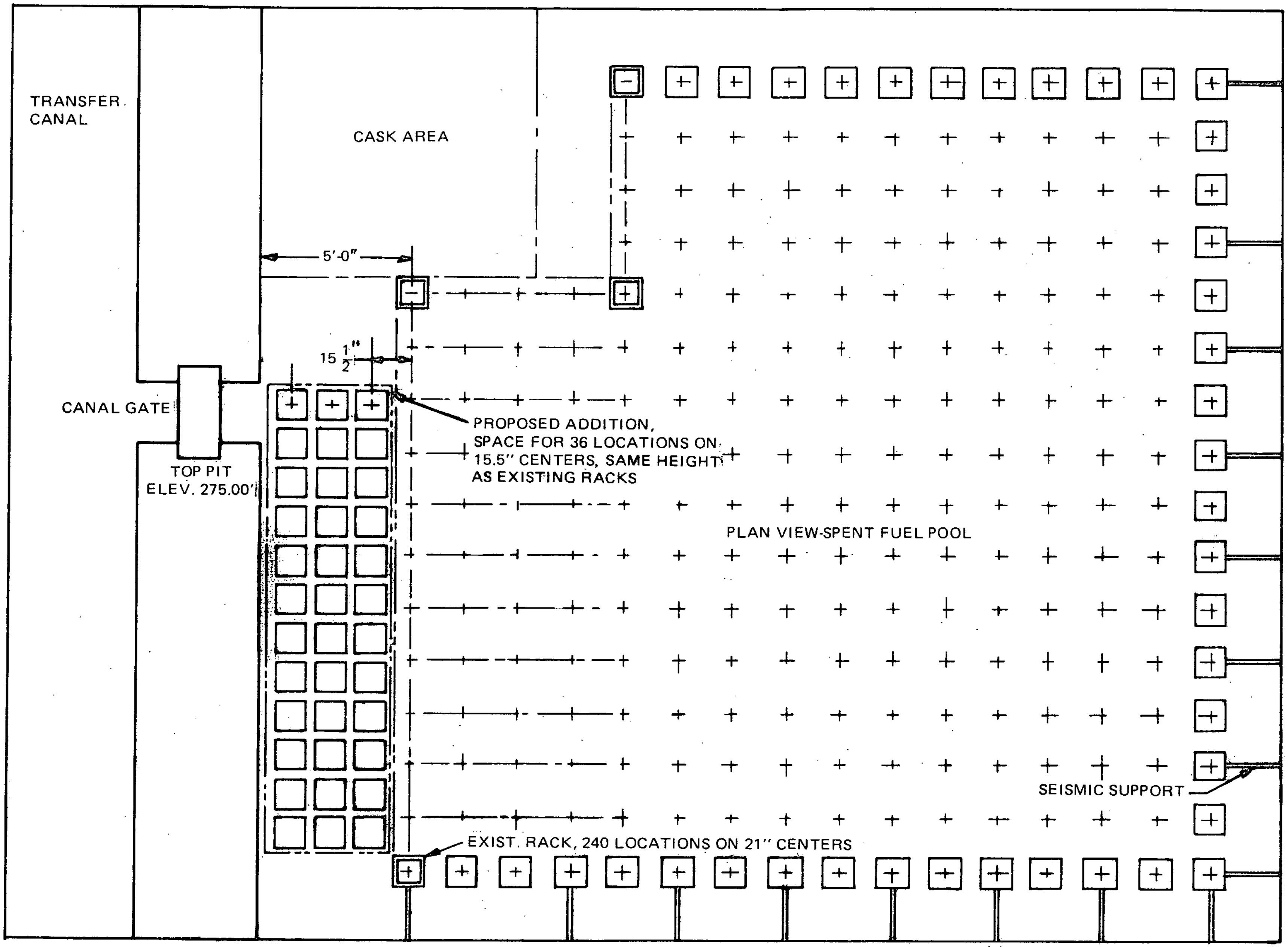


Figure 2.1–1 LOCATION OF NEW RACKS IN SPENT FUEL STORAGE PIT

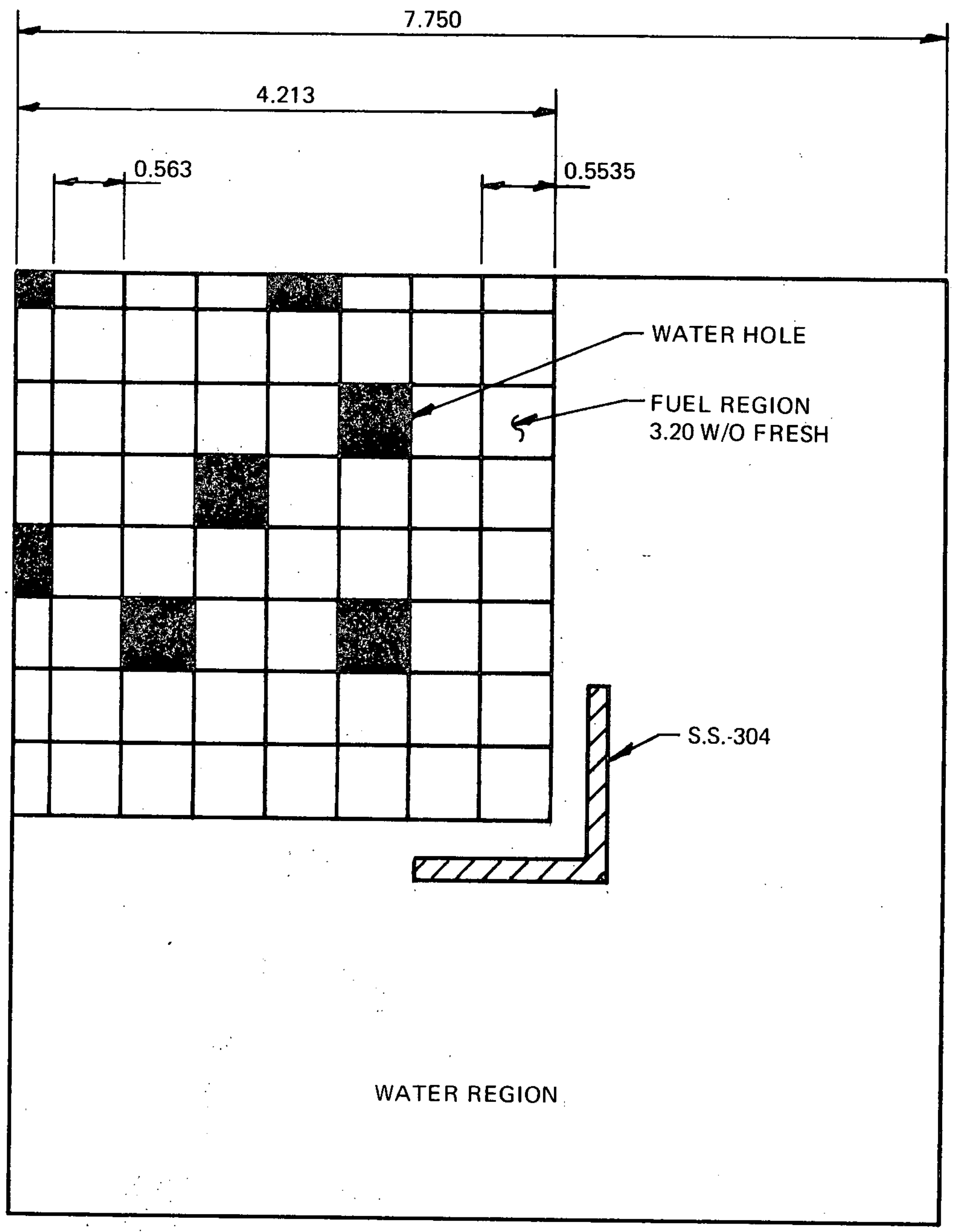


FIGURE 3.1–1 CROSS–SECTION OF FUEL/RACK GEOMETRY ANALYZED

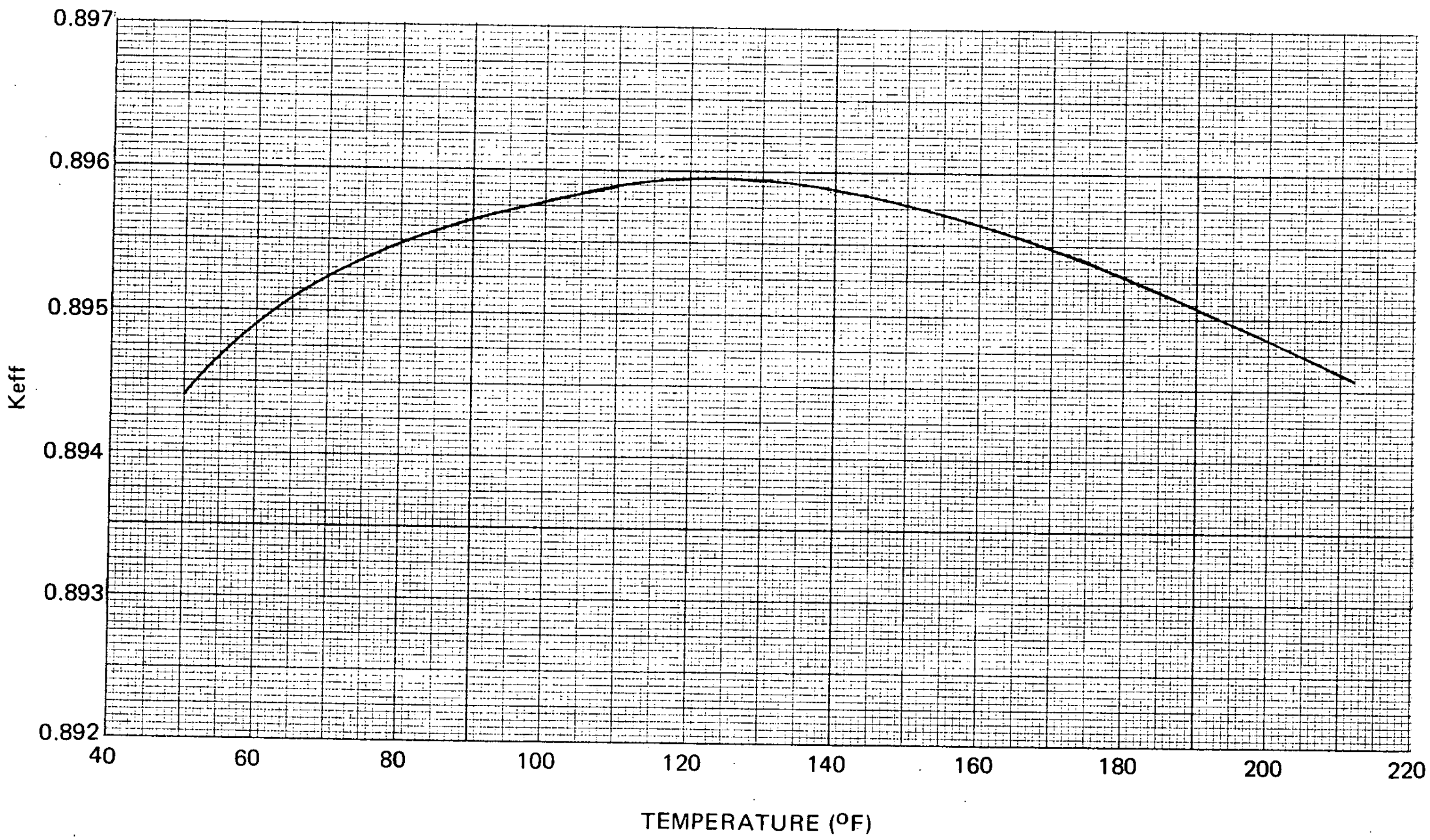


FIGURE 3.1–2 EFFECTIVE NEUTRON MULTIPLICATION FACTOR VS. TEMPERATURE 15.5" CENTER–TO–CENTER SPACING–NO BORON

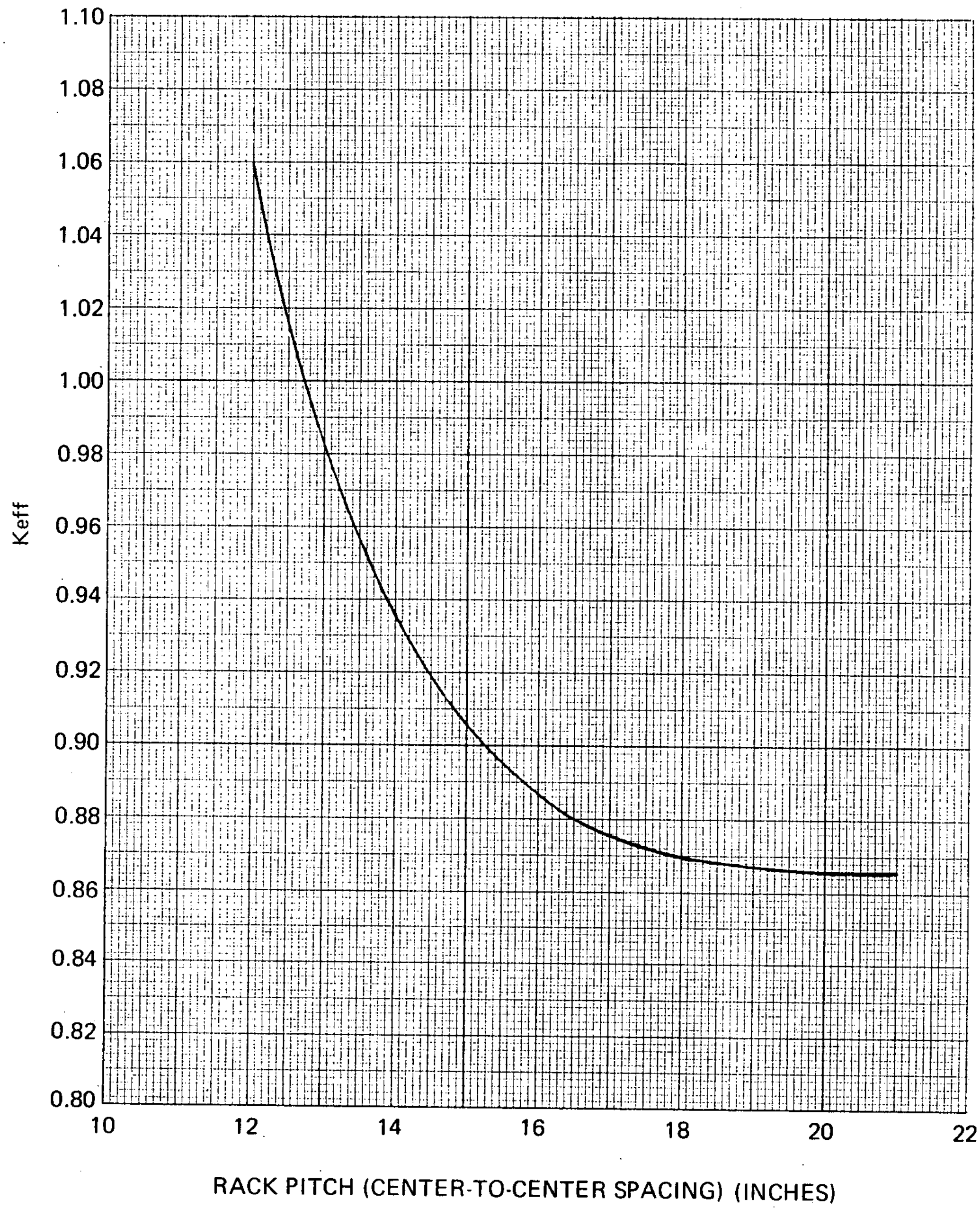


FIGURE 3.1–3 EFFECTIVE NUETRON MULTIPLICATION FACTOR VS. RACK PITCH FUEL AND MODERATOR TEMP. OF 125° F, NO BORON

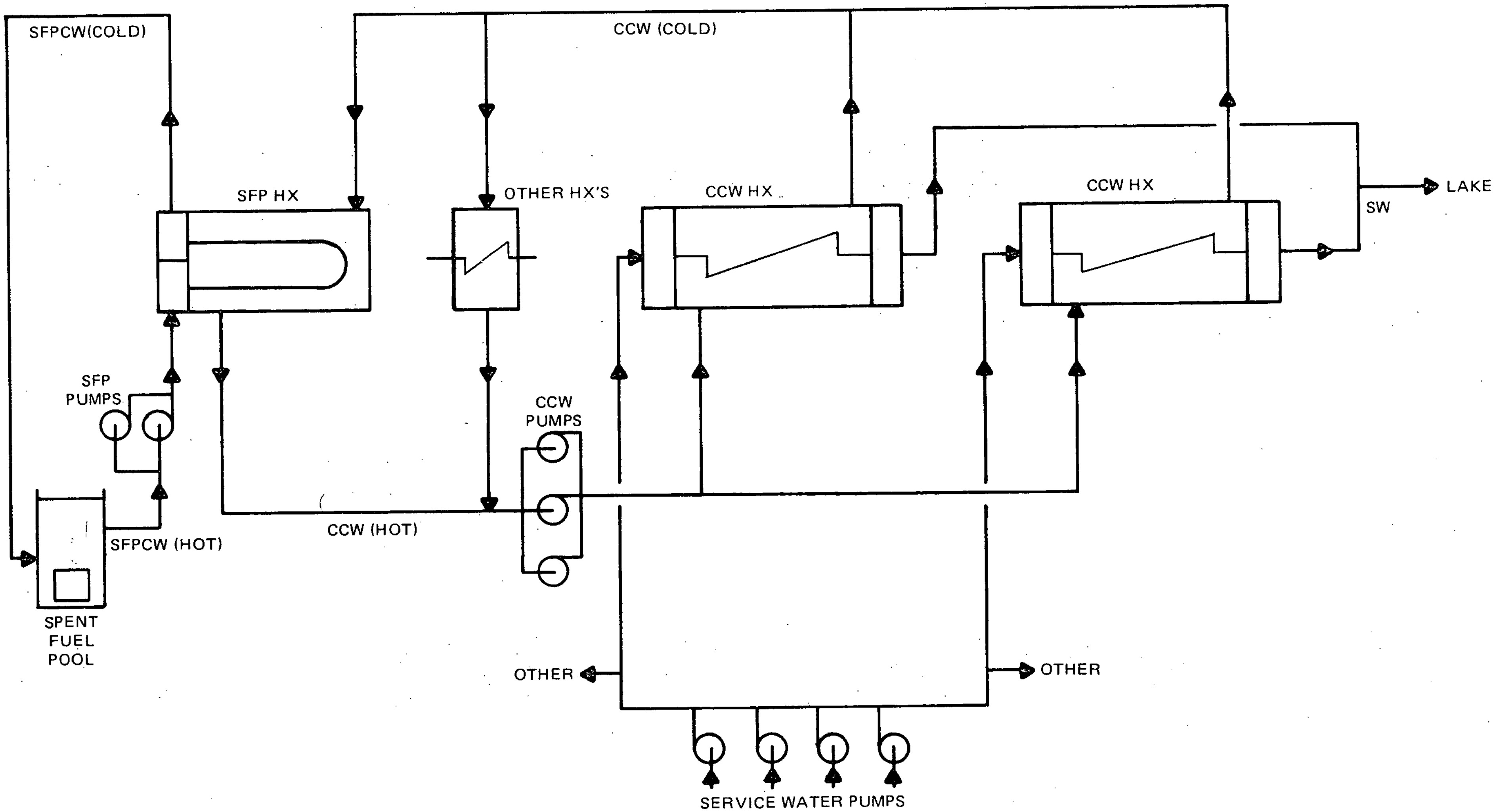


Figure 3.3–1 AUXILIARY COOLING SYSTEMS—FLOW DIAGRAM

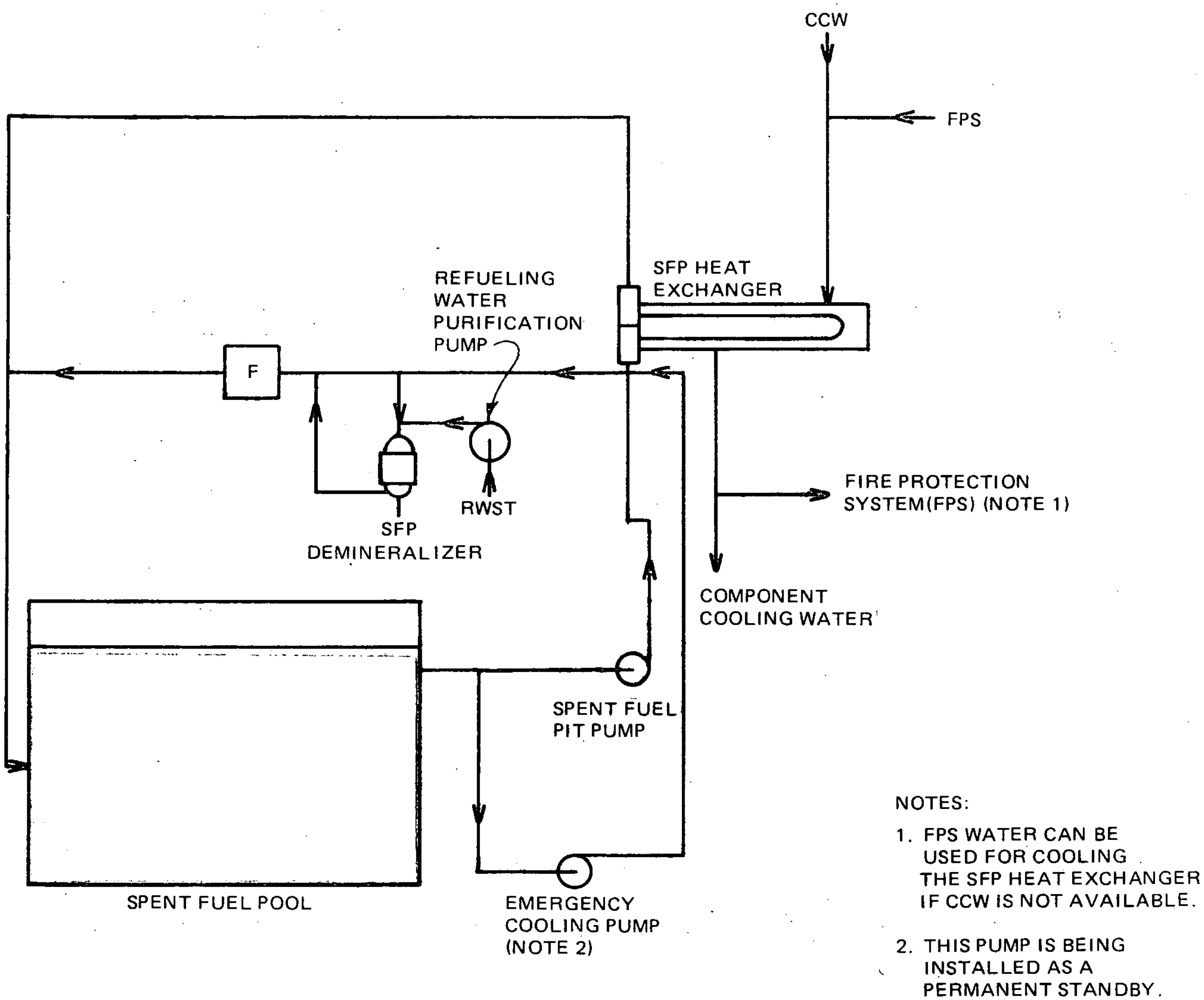
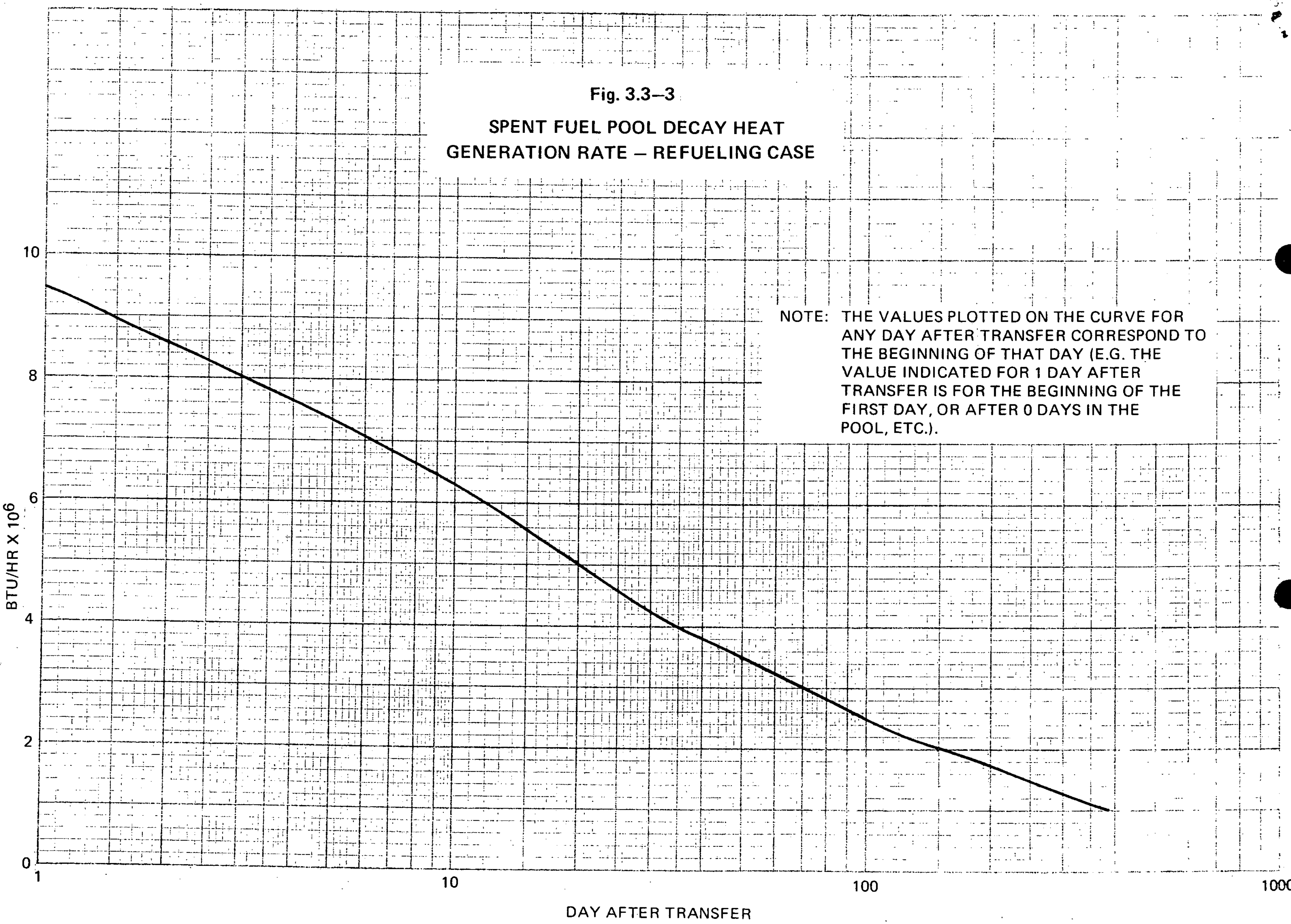


Figure 3.3–2 SPENT FUEL POOL COOLING SYSTEM—FLOW DIAGRAM

Fig. 3.3–3

SPENT FUEL POOL DECAY HEAT GENERATION RATE – REFUELING CASE



NOTE: THE VALUES PLOTTED ON THE CURVE FOR ANY DAY AFTER TRANSFER CORRESPOND TO THE BEGINNING OF THAT DAY (E.G. THE VALUE INDICATED FOR 1 DAY AFTER TRANSFER IS FOR THE BEGINNING OF THE FIRST DAY, OR AFTER 0 DAYS IN THE POOL, ETC.).

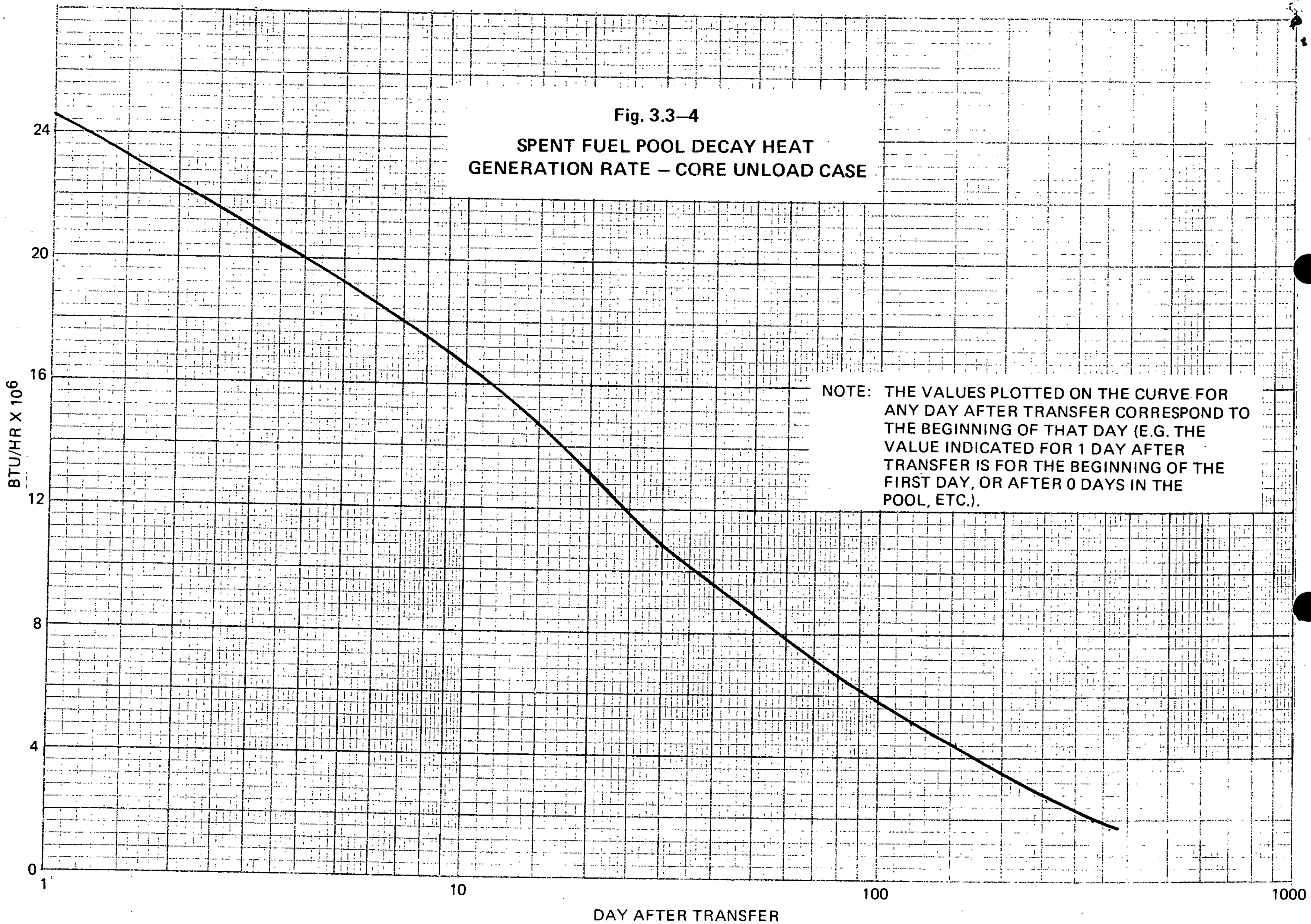


Fig. 3.3–4
SPENT FUEL POOL DECAY HEAT GENERATION RATE – CORE UNLOAD CASE
NOTE: THE VALUES PLOTTED ON THE CURVE FOR ANY DAY AFTER TRANSFER CORRESPOND TO THE BEGINNING OF THAT DAY (E.G. THE VALUE INDICATED FOR 1 DAY AFTER TRANSFER IS FOR THE BEGINNING OF THE FIRST DAY, OR AFTER 0 DAYS IN THE POOL, ETC.).
BTU/HR X 10^6
24
20
16
12
8
4
0
1
10
100
1000
DAY AFTER TRANSFER