OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION UNIT No. 1

SPENT FUEL STORAGE

RACK MODIFICATIONS

JANUARY 1983

Revision 1

Rev. 1   Date 01/21/83

A COMPARISON OF THE OPPD LICENSING SUBMITTAL
TO THE NRC'S OT POSITION FOR REVIEW AND ACCEPTANCE OF
SPENT FUEL POOL STORAGE AND HANDLING APPLICATIONS
APRIL 14, 1978, REVISION JANUARY 18, 1979

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*The proposed modification meets the intent of the NRC position paper.

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1.0 INTRODUCTION AND BACKGROUND

In early 1977, the Federal Government announced the indefinite suspension of the reprocessing of spent nuclear fuel to recover the unused fissile material. In 1977, the government also announced plans to acquire facilities for receipt and storage of spent fuel assemblies from light water reactors, but such plans have neither been implemented nor authorized. Therefore, utilities with operating reactors have had to make other provisions for storing spent fuel discharged from such reactors.

Even prior to these federal actions in 1977, it became obvious that there would not be adequate reprocessing capacity available on a schedule sufficient to handle all the spent fuel then being discharged from operating reactors. This shortfall in capacity was the result of one nearly complete reprocessing facility being declared commercially impractical and two other facilities being indefinitely delayed due to changes in federal regulations and other licensing difficulties.

In 1975, OPPD responded to this apparent shortfall in available reprocessing capacity with a decision to replace the original fuel storage racks in the spent fuel pool. In 1976, the original spent fuel racks were replaced with higher density storage racks which would provide adequate storage space on-site in the spent fuel pool for all fuel to be discharged through 1985. Since this decision had to be made well in advance of the repocessing deferral announced in 1977, providing racks with storage capacity to handle fuel until 1985 appeared to be a prudent decision which was generally consistent with industry practice at that time.

However, the subsequent failure of the Federal government to assume responsibility for spent fuel storage in lieu of reprocessing has made it necessary for OPPD to once again consider other available means for providing storage for spent fuel to be discharged from Fort Calhoun beyond 1985.

OPPD conducted a detailed evaluation of the alternatives available for disposition of spent fuel to be discharged from Fort Calhoun as summarized in Section 2.0 of this document. Based on that evaluation, it was concluded that reracking the Fort Calhoun spent fuel pool again with higher density storage racks which take credit for fuel burn-up, could provide a maximum capacity to accommodate all spent fuel assemblies to be discharged through 1994. Further, it was also established that such racks could be designed to accommodate storage canisters which could be filled with fuel rods which have been disassembled and tightly packed in the canister. This procedure allows the equivalent of two fuel assemblies to be stored in a single storage rack position and can thereby provide sufficient storage capacity to accommodate all the spent fuel likely to be discharged from Fort Calhoun during its useful operating lifetime. Further, as described in Section 10.0 of this document, these evaluations have shown that reracking the spent fuel pool is also by far the least expensive alternative for accommodating the spent fuel discharged from Fort Calhoun. Therefore, OPPD chose to proceed with the design, licensing, procurement and installation of replacement spent fuel racks because it was the only viable alternative available.

2.0 EVALUATION OF ALTERNATIVES FOR DISPOSTION OF SPENT FUEL TO BE DISCHARGED FROM FORT CALHOUN

The alternatives for spent fuel disposition considered by OPPD are shown in Figure 2.0-1, which also summarizes the criteria used for the evaluation of these alternatives. In view of the history of recurrent spent fuel disposition problems, a solution was sought which would accommodate all the spent fuel likely to be discharged from Fort Calhoun until the expiration of its operating license in 2008. Table 2.0-1 provides a tentative schedule of fuel discharges from Fort Calhoun that was used as the basis in this evaluation.

One of the alternatives considered was the potential for reduction of spent fuel storage requirements by reducing the number of fuel assemblies discharged. While fuel cycle optimization might allow some reduction in accumulative yearly discharges, such potential reductions are not of sufficient magnitude to significantly affect the solution to the spent fuel disposition problem for Fort Calhoun.

Reprocessing prospects, both domestic and foreign, were considered in the evaluation, but the present uncertainties surrounding the reprocessing option precluded it from being a viable solution.

Serious consideration was given to the option involving stand alone independent spent fuel storage facilities (ISFSF). Currently, the only viable option involving an ISFSF would be for OPPD to construct a ISFSF either on-site or off-site. However, this option was found to be one of the most costly alternatives and also involved significant licensing risks which could delay implementation. Therefore, use of ISFSF was determined to be an undesirable alternative.

Several on site storage options which do not involve modifications to the present spent fuel pool or racks were carefully considered during the evaluation. One or more of these options appear to offer a storage alternative which may be less costly and could involve fewer licensing risks than utilization of an ISFSF. In addition, there appear to be no serious technical problems associated with implementation of most of these concepts. However, these storage concepts have yet to be demonstrated or licensed for use at any nuclear power plant in the United States. In addition, these storage concepts are significantly more costly than alternatives involving the use of existing on-site facilities.

In consideration of the alternatives involving the use of existing on-site facilities, it was determined that replacing the presently installed stainless steel racks with efficient high density racks, which utilize two region storage, would be the best alternative. Region 1 utilizes poison material, and Region 2 takes credit for fuel burnup. This could extend the date for loss of full core discharge capability from 1985 to 1994. It was also determined that by implementing disassembly and compact storage (DCS), and taking credit for depleted fuel of individual fuel rods in canisters, the new high density racks could easily accommodate all the fuel likely to be discharged from Fort Calhoun through the year 2008. In addition, reracking with high density racks utilizing a poison material in Region 1 was determined to be among the least costly alternatives and to involve the lowest licensing risks. This region is reserved for full core discharge, (FCD) and damaged fuel. Region 2 takes credit for fuel burn-up, and contains no poison material. Damaged fuel may also be stored in this region if the fuel burnup meets the criteria listed in Section 4.0. Although DCS has not yet been approved for use in a commercial nuclear power plant, it was determined that there are no known technical problems barring implementation of this space saving concept.

However, the timing for demonstration, approval, and implementation of actual DCS operations involves significant uncertainty, and the on-site alternatives which require early implementation of DCS thereby share this uncertainty.

Such alternatives would have utilized DCS in combination with the presently installed racks (to the extent consistent with structural and seismic considerations) or with a new and more efficient non-poison rack design. While the alternatives which rely on early implementation of DCS are among the least costly, the inherent schedule uncertainty associated with this alternative is undesirable.

Therefore, the favored alternative resulting from our evaluation is the replacement of the current racks with higher density racks which utilize a neutron poison material in Region 1 and no neutron poison material in Region 2. These new racks would expand the fuel pool storage capacity to 729 whole fuel assemblies, as compared to the current capacity of 483 assemblies. Region 1 contains 198 cells, and Region 2 contains 531 cells. If it becomes necessary to implement DCS operations in 1994, the ultimate storage capacity of the racks with DCS will be approximately 1325 fuel assemblies, which is in excess of the current estimate for total storage requirements for Fort Calhoun.

TABLE 2.0-1

Fort Calhoun Unit No. 1

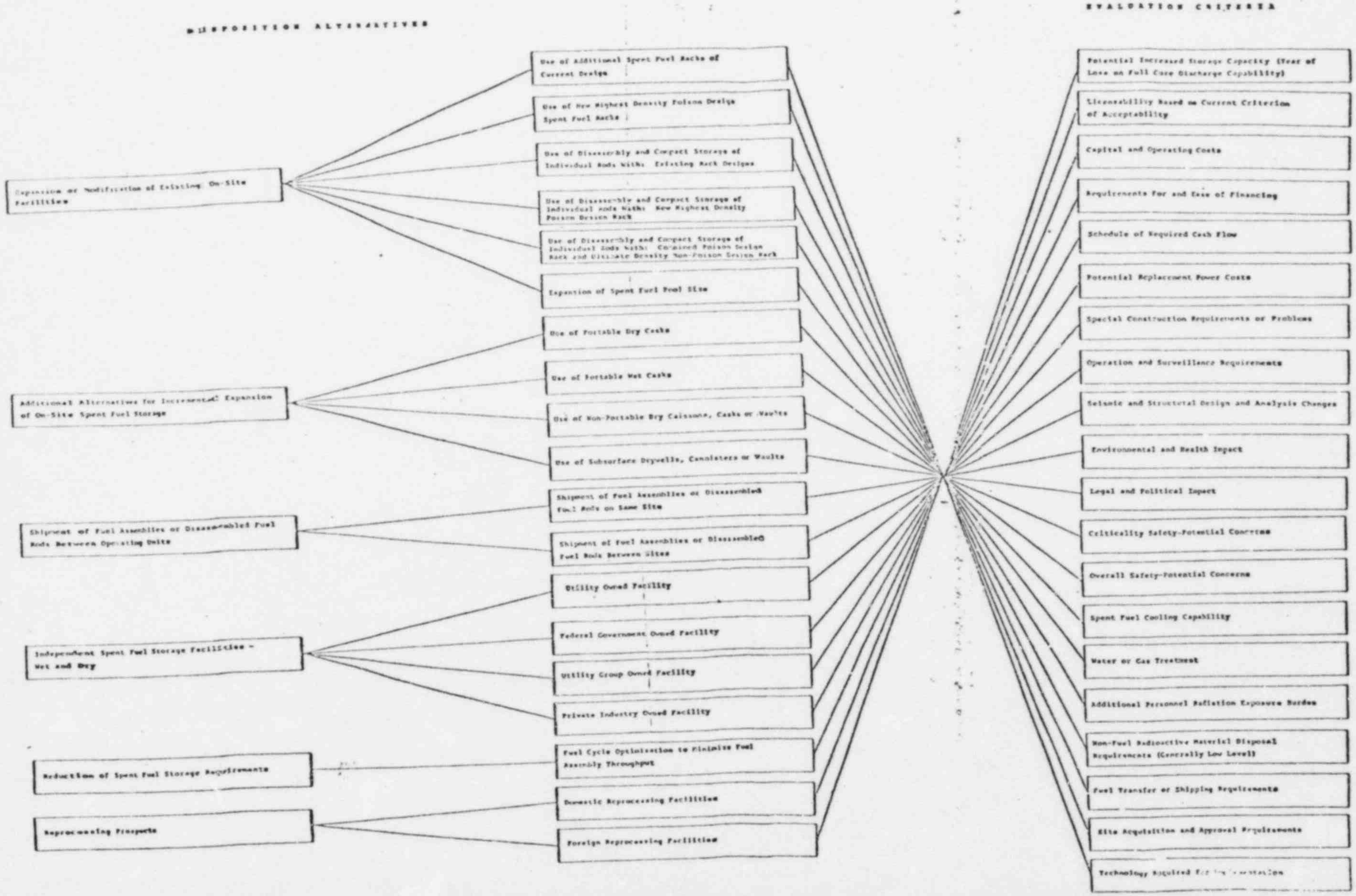
Spent Fuel Discharge Schedule

(No. of Assemblies)

Cycle	Refueling Outage Start	End	Number of Assemblies Discharged	Cumulative Number Of Assemblies Discharged
5	01/80	- 06/80	40	197
6	09/81	- 11/81	40	237
7	12/82	- 03/83	28	265
8	03/84	- 05/84	44	309
9	09/85	- 11/85	45	354
10	03/87	- 05/87	44	398
11	09/88	- 11/88	44	442
12	03/90	- 05/90	45	487
13	09/91	- 11/91	44	531
14	02/93	- 05/93	44	575
15	09/94	- 11/94	45	620
16	09/96	- 05/96	44	664
17	09/97	- 11/97	44	708
18	03/99	- 05/99	45	753
19	09/2000	- 11/2000	44	797
20	03/2002	- 05/2002	44	841
21	09/2003	- 12/2003	45	886
22	03/2005	- 05/2005	44	930
23	09/2006	- 11/2006	44	974
24	03/2008	- 05/2008	45	1019
25	09/2009	- 11/2009	44	1063
26	03/2011	- 05/2011	44	1107
27	09/2012	- 11/2012	45	1152

FIGURE 2.0-1

ALTERNATIVES FOR SPENT FUEL DISPOSITION AND CRITERIA FOR EVALUATION



3.0 GENERAL DESCRIPTION OF THE NEW SPENT FUEL STORAGE RACKS

The new spent fuel storage racks will be made up of two regions. Region 1 will comprise fuel racks containing Boraflex^R. These racks will contain 198 fuel cells to be used for a Full Core Discharge, and for failed or damaged fuel. This Region will also be used for temporary storage of freshly discharged fuel. Region 2 racks with 531 fuel cells will not contain any poison material, and as such credit is taken for depleted fuel to be stored in this region. Region 2 can also store damaged fuel as long as the fuel depletion is such that it meets the criteria of Section 4.0. Administrative controls and procedures will be established to limit storage of freshly discharged fuel in Region 1 only. Also during refueling operation mechanical interlocks will be provided to limit travel of the spent fuel pool fuel handling machine (FH-12) to Region 1 only. This will help to prevent any error which might result in the movement of a fuel assembly with less than the required burnup to Region 2. A more detailed discussion of this storage concept is described in the Criticality Analysis Section 4.0. Proposed administrative controls are discusssed in Sub-Section 3.7.

The proposed rack modules will be free standing and thus free to slide or rock on the floor during a seismic event. The sliding and rocking analyses are discussed in Section 7.0.

Free standing racks offer the following advantages:

1) Uplift loads are eliminated on any pool floor embedments,

2) Horizontal forces are reduced relative to a vertically restrained rack,

3) All modules are designed to be independently self-supporting, and

4) Individual modules can be easily removed and installed.

3.1 MODULE CONSTRUCTION

Except for dimensions these racks are structured and fabricated similar to the Alabamba Power Company, Farley's racks, which were licensed by the NRC. The rack module is composed of poison canisters, a bottom grid, and rack support feet, (Ref. PaR dwg P-33929-D). Except for the neutron absorber and the threaded feet all rack materials are fabricated from 300 series stainless steel.

Poison canisters are die formed and welded together at the top to form the top grid. These canisters also provide lead-in surfaces for the fuel.

The poison canisters are also die formed and welded to the bottom grid. The fuel support surfaces and fuel rack support feet are integral to the bottom grid. The feet can be adjusted to facilitate leveling at installation.

The neutron absorber (Boraflex®) used in Region 1 is encased with a stainless steel wrapper and is firmly held against the inner canister, as shown on PaR drawing No. P-33929-D. This outer wrapper is composed of two 'L' shaped sheets, of .024" ± .002" stainless steel. These 'L' sheets are firmly pressed against the neutron absorber and inner canister and are then spot welded along the canister length at diagonally opposite corners.

The 'L' sheets have folded edges (double thickness) where they are to be joined to each other, to dissipate weld heat. The 'L' sheets are also formed at right angles to the canister length to prevent vertical movement of the neutron absorber. Pool water is free to come in contact with the neutron absorbers. The outer stainless steel sheets are to maintain the neutron absorber in position and not used as structural elements.

In this arrangement, a water space is provided between the Boraflex® sheets to produce a flux trap effect necessary to meet criticality control requirements. The close spacing of adjacent fuel assemblies in this design is made possible by minimizing the stainless steel and water which separates the edge of the fuel assemblies from the Boraflex® sheets.

This efficient design allows a minimum pitch between adjacent stored fuel assemblies of 9.90", with adjacent storage boxes spaced apart by 1.23" minimum. The storage box wall is .09" ± .004" thick. The Boraflex® sheets are .085" ± .010" thick with a minimum loading of .020 gm B^{10}/cm^2. Each storage cell is 161" long to accommodate the fuel assemblies which are 148.83".

The nominal and minimum interior square width dimension of the fuel can will be 8.46 inches and 8.43 inches, respectively. The minimum can opening will be checked by a full length gage with an 8.315" square cross section.

After assembly, each poison canister will be checked to assure its capability to accept a full length dummy fuel bundle (8.22" square cross section) vertically within +/-.125". This procedure will account for the combined cross sectional tolerance, straightness, twist, and opening squareness.

Each storage box will have a welded-in bottom cruciform. This bottom cruciform provides the vertical support for the fuel assembly, and the open bottom permits cooling water to flow freely up through the fuel assembly.

Module Size	Capacity	Quantity	Estimated Module Weight (lbs)	Total Cavities
8x9	72	1	15,800	72
7x9	63	7	13,825	441
6x9	54	4	11,350	216
			Total	729

3.2 STRUCTURAL MATERIALS

The following table lists the materials to be used in the rack modules.

All stainless steel property values are per the ASME Pressure Vessel Code, Section III, and Subsection NF, Appendix I. All material, fabrication and welding and inspection of the racks will be done in accordance with ASME Section III Subsection NF.

DESCRIPTION	ALLOY	MAT'L SPEC.	FY, MIN. (KSI)
Grid Members	304SS	ASTM A-240	30
Inner Canister	304SS	ASTM A-666 Grade B*	45
Outer Canister Wrapper	304SS	ASTM A-240	30

DESCRIPTION	ALLOY	MAT'L SPEC.	FY, MIN. (KSI)
Threaded Foot	17-4 PH H-1100	ASTM A-564	115
Foot Metal	304L	ASTM A-240 or A-276	30 **
Weld Filler Metal	308L		

*The ASTM A-666 material that is used is equivalently qualified to the ASME Code, Subsection NF (SA-240 material) with the exception of the higher yield stress (Grade B) which is documented by certified material test reports.

**The 304L Stainless Steel that will be ordered will have a minimum yield of 30 KSI.

The allowable stresses in the heat affected zones of the weld and base material, utilizing ASTM A-666 in storage rack construction, are higher than the allowables of ASME SA-240 steel. Neither the ASME or ASTM Code requires an assured reduction in yield strength in heat affected zones of the weld and base material. Tests have been performed by an independent testing laboratory on samples of welded A-666, Grade B material. These tests indicate an average yield stress 17.9 percent higher than the minimum yield stress specified by ASTM for unwelded A-666 Grade B material and 76.9 percent higher than the minimum yield stress specified for SA-240 material. Further, A-666 material is currently in use in a comparable spent fuel rack design (Alabama Power Company, J. M. Farley Plant). This design was licensed by the NRC and has design stresses in the heat affected zones based upon the ASTM Code allowable stresses for the A-666 material. The design stresses for the proposed Fort Calhoun racks will be comparable to those currently licensed.

In this particular design the stress corrosion cracking problem is not a concern due to several factors. Stress corrosion cracking may occur where extensive intergranular precipitation is found in combination with tensile stresses. While some residual stresses cannot be avoided in this design, the amount of heat input from welding is small and the rate of cooling rapid. Low heat input, a maximum interpass temperature of 350°F for all welds, and rapid cooling due to the configuration minimizes the intergranular precipitation. Intergranular precipitation occurs when a high temperature is held for a long period of time: i.e., for 2 hours, which will not be the case.

In addition to the above factors all welds will be made with 308L filler metal and the heaviest weldment (the foot corner) will have a .03 percent maximum carbon content. These factors also greatly reduce the level of susceptibililty to stress corrosion.

The racks will be fabricated and welded in accordance with ASME Section III, Subsection NF.

Spent fuel storage modules have been manufactured previously of welded Type 304 stainless steel for nuclear applications with no indication of stress corrosion cracking problems.

Rack verticality is achieved by remotely adjusting the pedestals. The pedestals are designed with leveling screws (refer to P-33929-D) that can be remotely adjusted with a long handled tool. The racks will have to be lifted approximately 1/4" by the auxiliary building crane to take the load off the pedestals. Each pedestal will be adjusted to achieve verticality. Use of the long handled tools will reduce exposure to personnel installing the racks. (refer to P-33931-D for tool)

Each rack pedestal bottom plate rests on a bearing plate which is sized to keep the floor loadings below 500 psi. This provides the mechanism for transmitting rack horizontal loads through the pedestal to the shim plate to the floor.

The leveling screw design will meet all the necessary criteria and requirements and will provide a safe method for installing the new fuel racks.

3.3 POISON MATERIAL EVALUATION

Boraflex® has been selected as the poison material for the racks based on the results of an extensive test program extending over the past several years. This material is a silicone rubber, boron carbide solid which is extremely stable in the spent fuel pool environment of hot borated water and ionizing radiation. A number of tests have been performed by the vendor to verify the suitability of this material for spent fuel racks and additional test programs are continuing.

Evaluation of the effects of gamma irradiation on Boraflex® in various environments has been conducted and is continuing at the University of Michigan. The reference listed below presents data which demonstrates that the exposure of Boraflex® in air to 2.81×10^8 rads gamma from a spent fuel source results in no significant physical changes and does not generate any harmful gases. Data is also presented showing that irradiation of 1.03×10^{11} rads gamma, with a substantial concurrent air neutron flux, deionized water, and borated water environments causes some increase in hardness and change of tensile strength of Boraflex®. During this irradiation a certain amount of gas is generated, but beyond approximately 1×10^{10} rads gamma, the rate of gas generation does not exceed the rate observed when a sample container filled with only borated or deionized water is irradiated.

To date, all test data indicate that, through a cumulative irradiation in excess of 1×10^{11} rads gamma, there is no significant deterioration of Boraflex® which could affect its suitability for long term use in the spent fuel racks. As of Nov. 1982, Boraflex® has been approved for use in the spent fuel racks at six operating nuclear plants.

Ref: "Irradiation Study of Boraflex Neutron Shielding Materials," Report 748-10-1, Brand Industries Services, Inc., July 25, 1979.

3.4 CAVITY IDENTIFICATION SYSTEM

A cavity identification system will be provided on the perimeter of the fuel storage racks. Stainless steel strips with alphanumeric characters (approx. 2.5"-3" size characters), which can be read at the top of the fuel pool, will be attached to the perimeter of the storage rack array, where space permits. This type of identification has been used and well accepted at numerous plants. Also cell numbers will be etched on the sloped lead-ins on one side of each cell. This identification will be readable through the under water camera, and may be used as a back up for verification/auditing purposes only.

3.5 INSTALLATION

At the time of reracking, the Fort Calhoun spent fuel pool will contain approximately 265 spent fuel assemblies, and therefore, careful consideration will be given to the sequence of rack removal and installation. Procedures will be established so that any crane lifts of either the replacement or existing racks will not be conducted over racks which contain spent fuel assemblies. Such procedures will ensure that a potential rack handling accident will not result in any significant release of radioactivity from the damage of spent fuel stored in the pool.

The use of special devices will comply with the criteria listed in NUREG-0612, in that, special lifting devices used for rack installation (see PaR dwg. P-33932-D) will meet the criteria of ANSI N14.6 - 1978. Non-special devices for the removal of the existing spent fuel storage racks and the installation of the new spent fuel storage racks presently are available within the District. These devices meet the criteria of ANSI B30.9-1971.

The removal sequence of the existing racks is dictated by the design utilized to interconnect these racks. The first rack to be removed must be B-1 in the southwest corner of the pool, and this will be followed by sequential removal of racks A-4 through A-1. New racks will then be installed. Next, existing racks B-2 and A-8 through A-5 will be removed after relocating any fuel assemblies stored in these racks. New racks will then be installed. Existing racks B-3 and A-12 through A-9 will be removed. Finally, existing racks C-1 through C-6 will be removed, and new racks will be installed to complete the re-racking. Sufficient storage capacity will be maintained at all times during this operation to accommodate the approximately 265 spent fuel assemblies expected to be in the pool at the time of re-racking, plus a full core discharge.

Prior to removing each spent fuel assembly from the existing racks to the new racks (Regions 1 or 2), each spent fuel assembly burnup will be obtained to verify that this fuel can be discharged to the nonpoisoned racks. The criteria for storing fuel in Region 2 is listed in Section 3.7.

3.6 SAFE LOAD PATH FOR REMOVAL AND INSTALLATION OF SPENT FUEL STORAGE RACKS

This section addresses the methods and assumptions used to determine the safe load path for the re-racking process. This is not the actual procedure, but is a preliminary guideline to indicate what will be included in the actual procedures. A preliminary procedure is provided in Appendix 'A'.

3.7 ADMINISTRATIVE CONTROLS

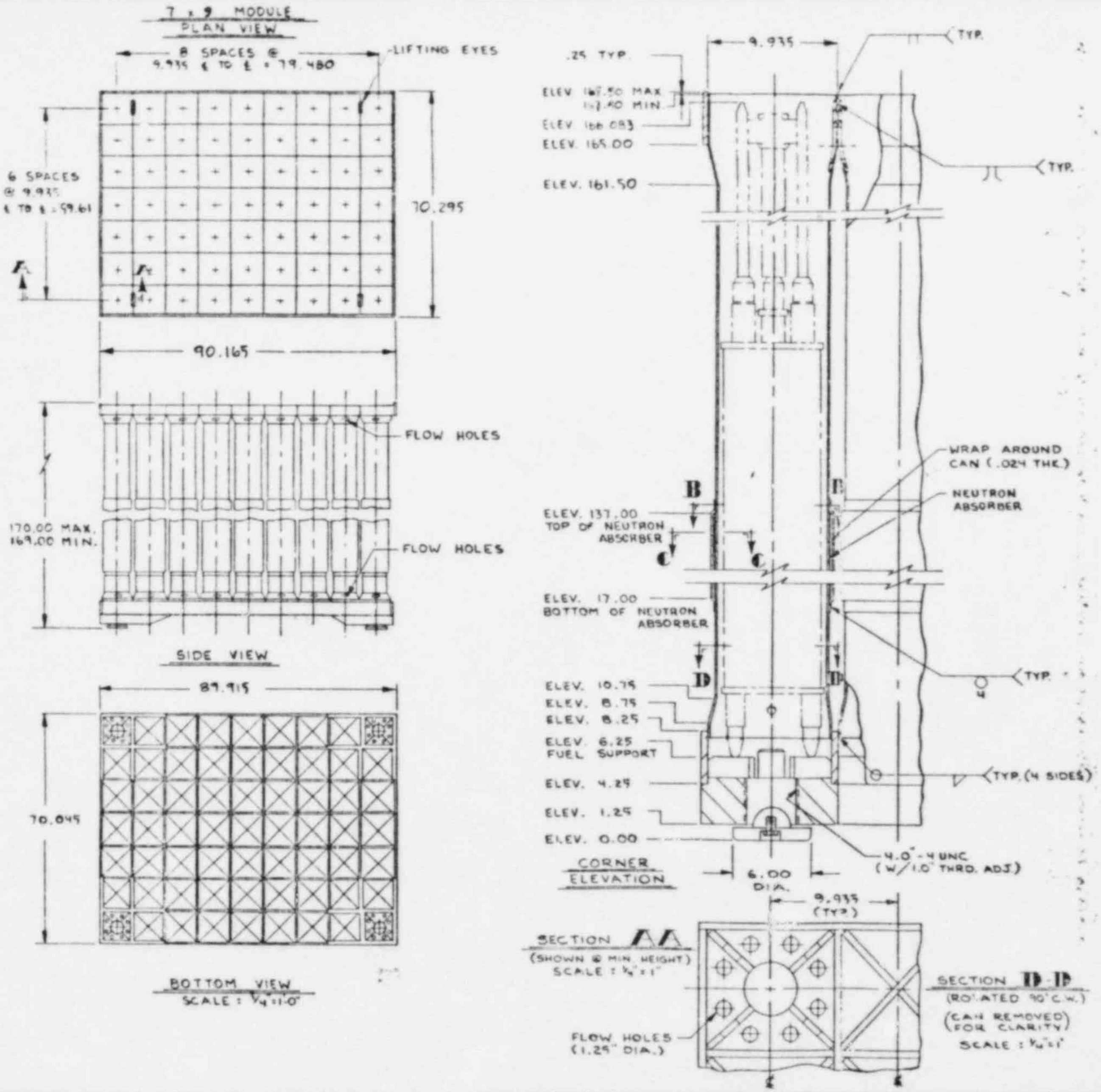
With a two region storage, administrative controls will be required to be met prior to discharging fuel to the spent fuel pool.

One of the controls will be that each fuel assembly burnup will be determined and verified after being discharged from the core. The burnup on spent fuel currently stored in the pool can be determined from records that exist. The information on fuel burnup is required to determine if the fuel should be stored in Region 1 or 2.

Fuel assembly burnup is currently calculated using the CECOR computer code which utilizes flux measurements from the incore detectors. The code then constructs power distributions and successive runs of the code on a core follow basis, provides the assembly burnups. At the end of a fuel cycle, a final exposure map and the burnup for all discharged assemblies can be obtained. The ROCS simulation code which is used for fuel management also provides calculations of the assembly burnups. The agreement between CECOR and ROCS is very good, so either may be used for determining the burnups of each discharged assembly.

Minimum assembly exposure will be verified and shown to be above the minimum required assembly exposure per Fig. 4.3-4.

Another control is that, during refueling conditions freshly discharged fuel from the reactor will be placed in Region 1. No freshly discharged fuel will be placed in Region 2 until minimum assembly exposure has been verified by calculation. To prevent freshly discharged fuel from being placed in Region 2, the fuel handling machine will be interlocked to prevent movement to that region during refueling operations. Full core discharges will be to Region 1 only.



NOTES:

1. ALL DIMENSIONS ARE REFERENCE.
2. RACK SHOWN IS A POISON RACK A NON-POISON RACK WILL BE SIMILAR EXCEPT IT WILL NOT HAVE ANY POISON OR OUTER WRAPPERS.

DEC 1 6 1982

REVISIONS					
ECO NO.	SYM	ZONE	DESCRIPTION	DATE	APPROVAL
	A		GENERAL REVISION		

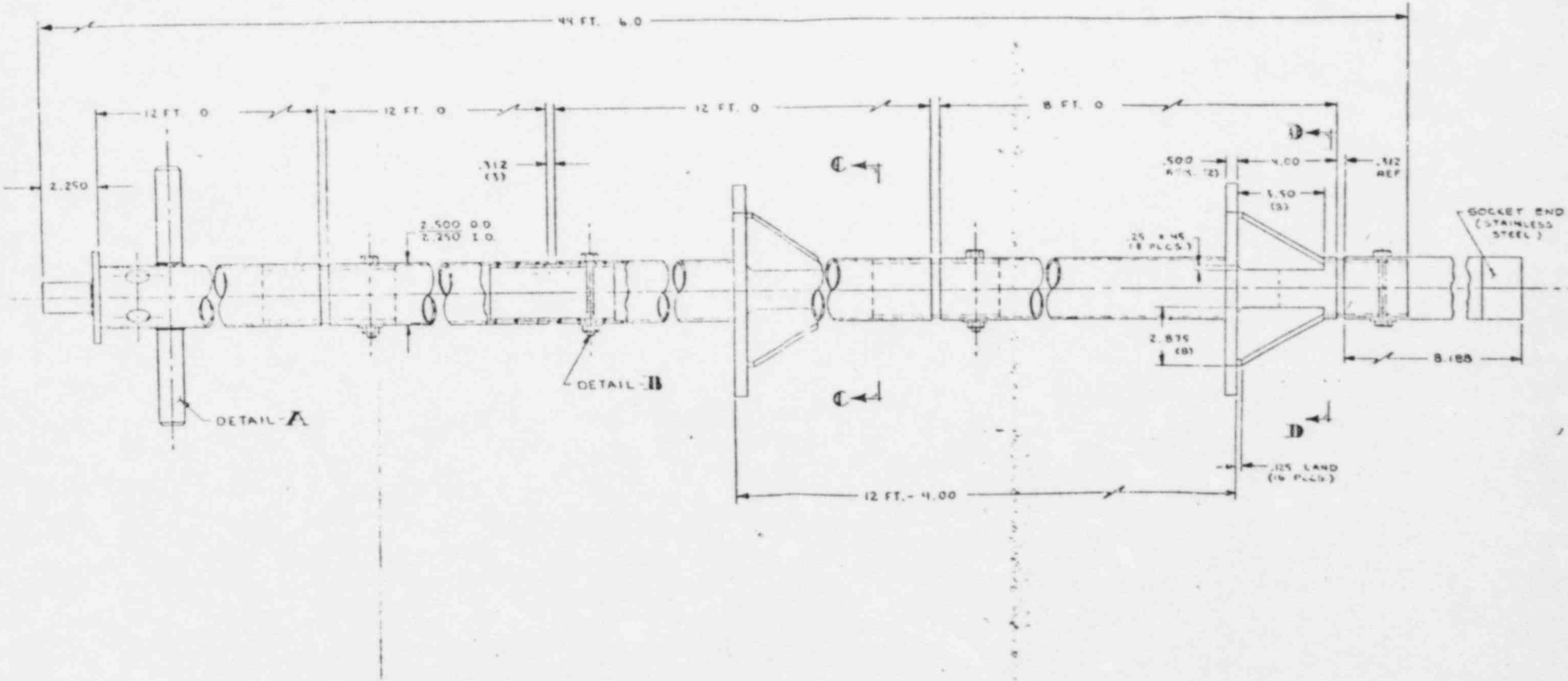
MATERIAL	STAINLESS STEEL
HEAT TREATMENT	—
FINISH	
DWN BY	8-25-82
CHKD BY	
ENGR	
APPD BY	
PROJ NO.	P. 1136-9
NEXT ASSY.	—
SCALE	NOTED
WT.	—

UNLESS OTHERWISE SPECIFIED: REMOVE BURRS AND SHARP EDGES OR STOCK FINISH. FILLETS AND RADII TO BE —. COUNTERSINK ALL TAPPED HOLES. TOLERANCES: [illegible]

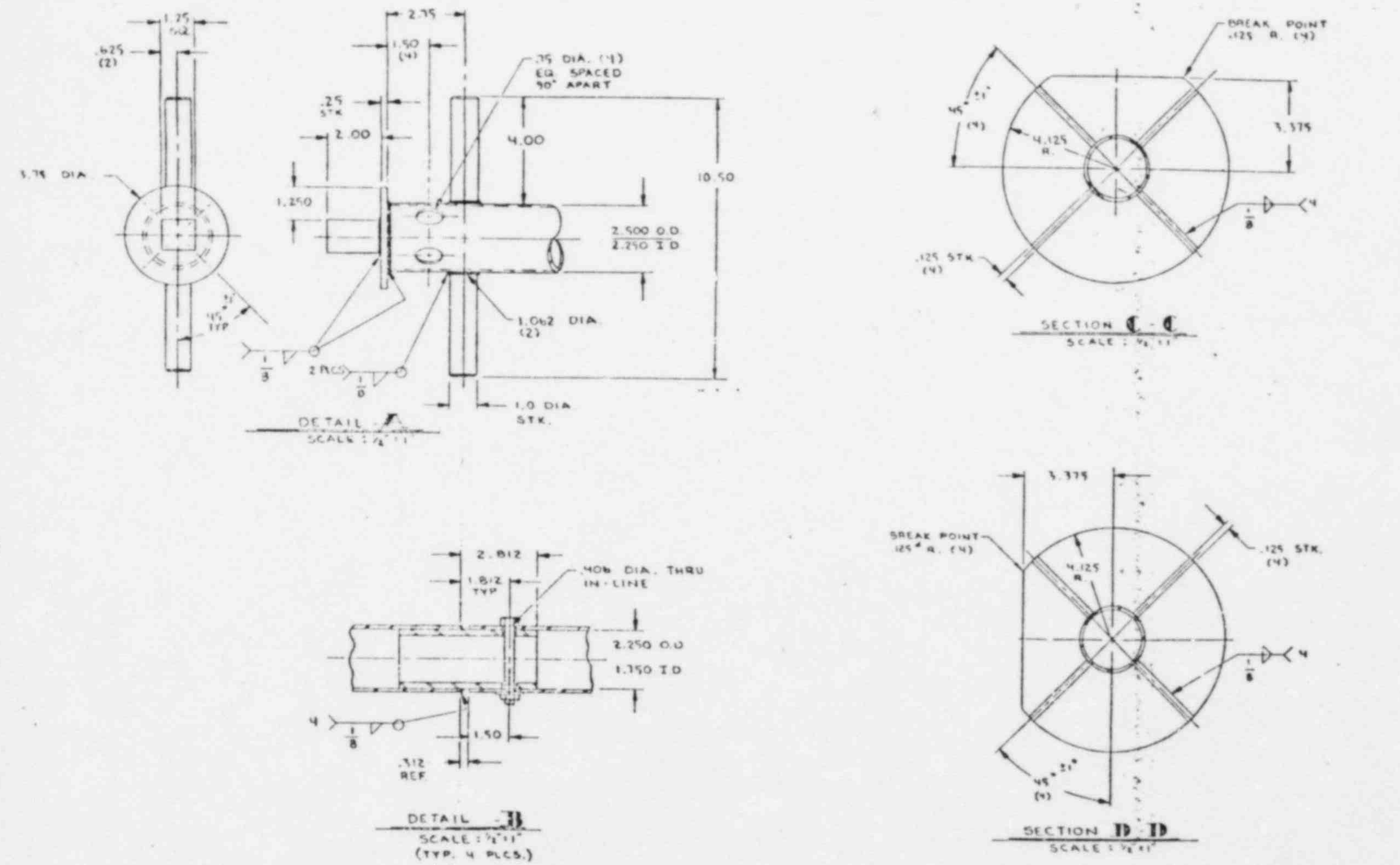
PaR Systems Corp.
St. Paul, Minn.

FORT CALHOUN
PROPOSED
SPENT FUEL MODULE

DWG NO. P-33923-D REV. [illegible] SH. 1 OF 1



MATERIAL ALUM. EXCEPT NOTED	DWN BY 8-26-82	PaR Systems Corp. St. Paul, Minn.
HEAT TREATMENT —	CHKD BY 9-3-82	
FINISH —	ENGR 9-82	
UNLESS OTHERWISE SPECIFIED REMOVE BURRS AND SHARP EDGES .125 OR STOCK FINISH FILLETS AND RADII TO BE — COUNTERSINK ALL TAPPED HOLES TOLERANCES	APPD BY	FORT CALHOUN LONG HANDLE TOOL ASSEMBLY
	PROJ NO. P-1136-9	
	NEXT ASSY. —	
	SCALE 1/2"=1 · WT. ~60	DWG NO. P-33931-D · REV · SH 1 OF 1



MATERIAL	—
HEAT TREATMENT	—
FINISH	—
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PaR Systems Corp.
St. Paul, Minn.

FORT CALHOUN
LONG HANDLE
TOOL ASSEMBLY

DWG NO. P-33931-D

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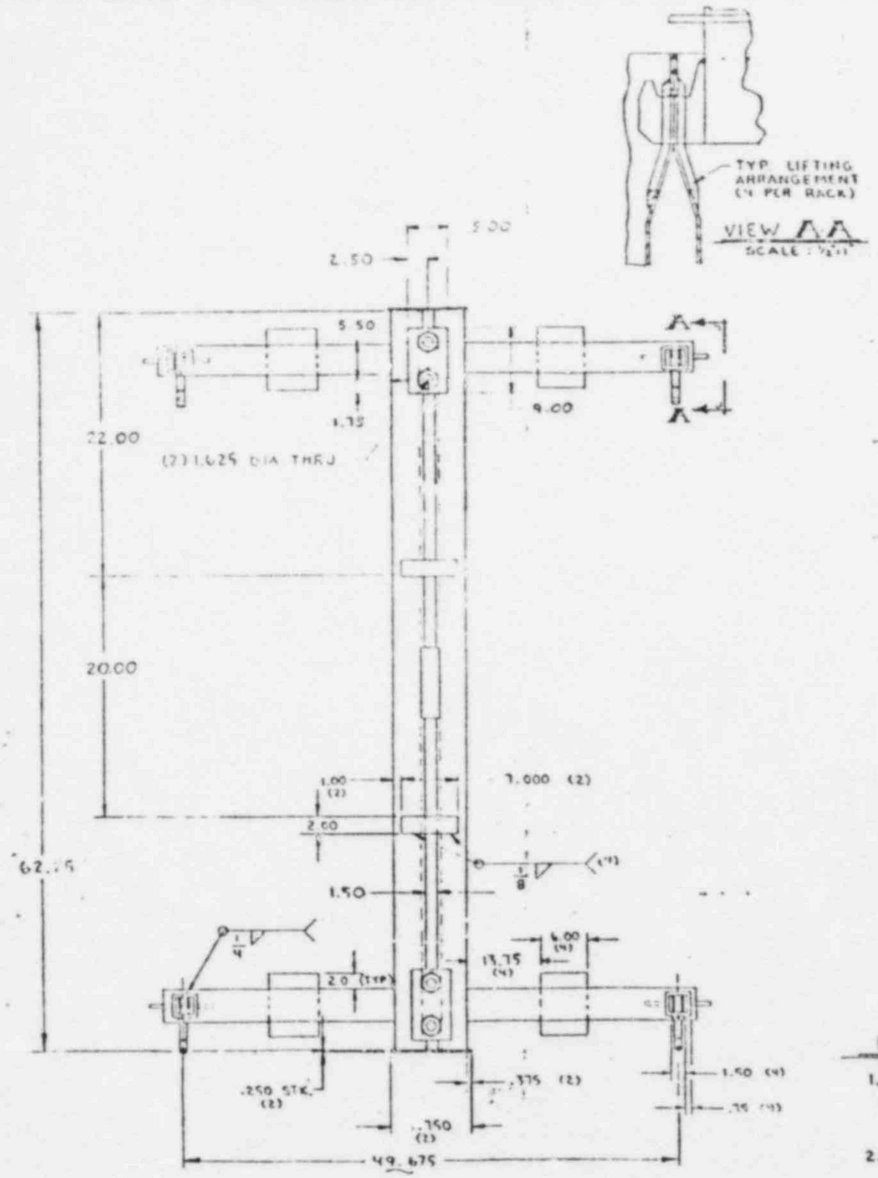

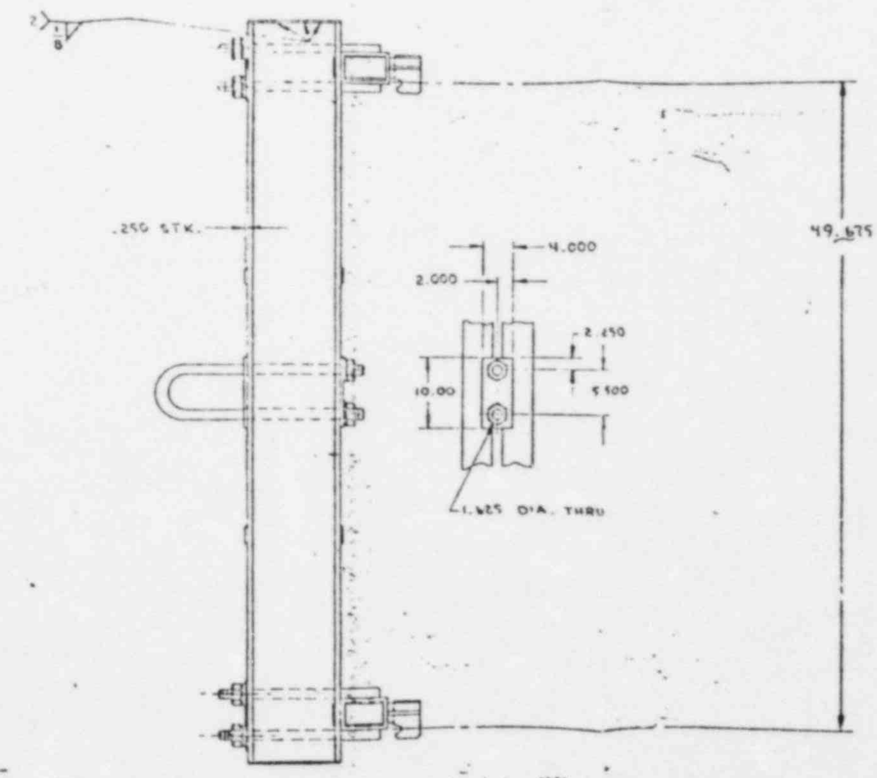

NOTES:

1. ALL ITEMS NOT S. STL. OR CAD PLATED CARBON STEEL, ARE TO BE PAINTED RUSTOLEUM GRAY.

2. FIXTURE SHOWN IS FOR 6 x 7 MODULE. THE FIXTURE FOR OTHER MODULES WOULD BE THE SAME, EXCEPT FOR SOME DIMENSIONAL CHANGES.

MATERIAL NOTED

HEAT TREATMENT —

FINISH —

UNLESS OTHERWISE SPECIFIED REMOVE BURRS AND SHARP EDGES

DWN BY 8-26-82

PROJ NO. P-1136-9

NEXT ASSY. —

PaR Systems Corp.
St. Paul, Minn.

FORT CALHOUN MODULE LIFTING FIXTURE

4.0 CRITICALITY ANALYSIS FOR THE SPENT FUEL STORAGE RACKS

The following discussion summarizes the design of the spent fuel racks with respect to the criticality design. The analytical technique described below is similar to that used to successfully license spent fuel racks for several other plants, the most recent being those for Point Beach Units 1 and 2, Nine Mile Point Unit 1, and Callaway.

During a normal refueling operation, each fuel assembly is first moved from the core to Region 1. After the refueling operation is complete and the suitability of each spent fuel assembly for movement into Region 2 is verified, this fuel may be moved into Region 2.

Region 1 of the pool is designed to maintain unirradiated fuel of up to 4.0 w/o U-235 in a safe, coolable, subcritical configuration. Region 2 is designed to store fuel which does not exceed pre-established reactivity criteria. Consequently, the limit on acceptable initial enrichment varies with the exposure at the time of storage. For instance, 4.0 w/o fuel is acceptable for storage in Region 2 only after a predetermined minimum exposure has been reached.

A somewhat lower minimum exposure would be acceptable for fuel with a lower initial enrichment. The criteria for Region 2 of the pool are specifically listed and compared to the present criteria in Table 4.0-1.

4.1 ANALYTICAL TECHNIQUE REGION 1 (Assumes Unirradiated Fuel)

The LEOPARD [1] computer program was used to generate macroscopic cross sections for input into four energy group diffusion theory calculations which were performed with the PDQ-7[2] program. LEOPARD calculates the neutron energy spectrum over the entire energy range from thermal to 10 Mev and determines averaged cross sections over appropriate energy groups. The fundamental methods used in the LEOPARD program are those used in the MUFT[3] and SOFOCATE[4] programs which were developed under the Naval Reactor Program and thus, are well founded and extensively tested techniques. In addition, Westinghouse Electric Corporation, the developers of the original LEOPARD program, demonstrated the accuracy of these methods by extensive analysis of measured critical assemblies consisting of slightly enriched UO_2 fuel rods.[5]

Pickard, Lowe and Garrick, Inc. (PLG) has made a number of improvements to the LEOPARD program to increase its accuracy for the calculation of reactivities in systems which contain significant amounts of plutonium mixed with UO_2. PLG has tested the accuracy of the program modifications by analyzing a series of UO_2 and PuO_2-UO_2 critical experiments. These benchmarking analyses not only demonstrate the improvements obtained for the analysis of PuO_2-UO_2 systems, but also demonstrate that these modifications have not adversely affected the accuracy of the PLG modified LEOPARD program for calculations of slightly enriched UO_2 systems.

The UO_2 criticality experiments chosen for benchmarking included variations in H_2O/UO_2 volume ratios, U-235 enrichments, pellet diameters and cladding materials. Although the LEOPARD model also accurately calculates the reactivity effects of soluble boron, these experiments have not been included in the LEOPARD benchmarking criticals since the spent fuel pool calculations do not involve soluble boron.

Neutron leakage was represented by using measured buckling input to infinite lattice LEOPARD calculations to represent the critical assembly. A summary of the results are provided in Table 4.1-1 for the 27 measured criticals chosen as being directly applicable for benchmarking the LEOPARD model for generating group average cross sections for spent fuel rack criticality calculations. The average calculated k_{eff} is 0.9979 and the standard deviation from this average is 0.0080 Δk. Reference 5 raised questions concerning the accuracy of the measured buckling reported for cases No. 12 through No. 19. If these cases are excluded, the average calculated k_{eff} for the remaining 19 experiments is 1.0006, with a standard deviation from this value of 0.0063 Δk. In all of these experiments there are significant uncertainties in the measured bucklings which are necessary inputs to the LEOPARD analysis. These uncertainties are of the same order of magnitude as the indicated errors in the LEOPARD results, and therefore, a more definitive set of experimental data was used to establish the accuracy of the combined LEOPARD/PDQ-7 model for the criticality analysis of the spent fuel racks.

The PDQ series of programs have been extensively developed and tested over a period of 20 years and the current version, PDQ-7, is an accurate and reliable model for calculation the subcritical margin of the proposed spent fuel rack arrangement. This code, or a mathematically equivalent method, is used by all U.S. suppliers of light water reactor cores and reload fuel. In addition, this code has received extensive utilization in the U.S. Naval Reactor Program.

As a specific demonstration of the accuracy of the calculational model used for the spent fuel rack calculations, the combined LEOPARD/PDQ-7 model was used to compare calculated critical parameters to measured parameters for fourteen just critical assemblies. The criticals are high neutron leakage systems with a large variation in U/H_2O volume ratio and include parameters in the same range as those applicable to the proposed fuel rack design. Experiments including soluble boron were included in this demonstration to ensure the ability of PDQ-7 to calculate neutron leakage effects. The use of soluble boron allows changes in the neutron leakage of the assembly while maintaining a uniform lattice and thus, allows a better test of the accuracy of the model. Furthermore, soluble boron eliminates the error associated with the measured bucklings which is inherent in the LEOPARD benchmarks, thus permitting determinations of the actual calculational uncertainty which must be accounted for in a spent fuel rack criticality analysis.

The combined LEOPARD/PDQ-7 calculations resulted in a calculated average k_{eff} of 0.9928 with a standard deviation of 0.0012 Δk. These results, as shown in Table 4.1-2 demonstrate that the LEOPARD/PDQ-7 calculational model can determine the reactivity of the proposed spent fuel rack arrangements with an accuracy of 0.010 Δk, at the 95 percent confidence level.

The cross sections for the Boraflex® neutron absorbing material which is an integral part of the design, were calculated using fundamental techniques that have been successfully applied in the past to thin heavily absorbing mediums such as control rods.

This procedure is straight forward and is comprised of several well defined steps:

1. The B^{10} from the thin Boraflex® sheets is homogenized in an appropriate amount of water and LEOPARD is used to obtain unshielded macroscopic B^{10} cross sections.

2. Integral transport theory is applied in slab geometry using They's method for calculating flux depressions and shielding factors to determine an appropriate B^{10} number density. This approach is similar to that of Amouyal and Benoist.

3. The B^{10} number density calculated in Step 2 is homogenized in water, and LEOPARD is used to obtain corrected macroscopic B^{10} cross sections.

4. Blackness theory is applied to obtain macroscopic cross sections which will produce the required boundary conditions at the surface of the Boraflex® sheets.

In addition to the fourteen critical assemblies in Table 4.1-2, the LEOPARD/PDQ-7 model was used to calculate the k_{eff} for twelve additional critical assemblies, seven of which incorporated thin, heavily-absorbing materials for which the procedure just described was used to determine the macroscopic cross section.

These twelve criticals were performed by Battelle Pacific Northwest Laboratories specifically for the purpose of providing benchmark critical experiments in support of spent fuel criticality analysis. They are described in detail in Reference 17. The results of these critical experiments are summarized in Table 4.1-3. The first seven of the twelve experiments included fixed neutron poison absorber plates, and the average calculated k_{eff} for these just critical assemblies was 0.9926, with a standard deviation of 0.0006 Δk. The other five critical experiments in this series do not include absorber plates and the average calculated k_{eff} for these critical assemblies was 0.9938, with standard deviation of 0.0014 Δk. The overall average calculated k_{eff} for the twelve assemblies was 0.9931, with a standard deviation value of 0.0011 Δk.

This extensive series of UO_2 critical experiments further supports the applicability of the methods described above for use in calculating the subcritical margin of the fuel storage rack design, and demonstrates that the calculational accuracy of 0.010 Δk at the 95 percent confidence level as established for the LEOPARD/PDQ-7 model, applies, equally well to designs incorporating fixed neutron absorbers for which blackness theory is used to calculate the macroscopic cross sections.

As a result of this approach to separately benchmark both the cross sections and the diffusion theory calculations against applicable critical assemblies, the "transport theory correction factor" is implicitly included in the derived calculational uncertainty factor.

The PDQ-7 program was used in the final predictions of the reactivity of the spent fuel storage racks. The calculations were performed in four energy groups and take into account all the significant geometric details of the fuel assemblies, fuel boxes, and major structural components. The geometry used for most of the calculations is a basic cell representing one-quarter of the area of a repeating array of stainless-steel boxes. The specific geometry of this basic cell is shown in Figure 4.1-1.

4.2 ANALYTICAL TECHNIQUE REGION 2 (Assumes Irradiated Fuel)

The following discussion is applicable to the fuel storage in Region 2 of the pool. Since the criticality analyses for Region 2 are subject to potentially larger uncertainties than those applicable to Region 1, the uncertainties applicable to the Region 2 analysis are independently derived.

The analytical methods used for criticality analysis of Region 1 are also incorporated into the criticality analysis of Region 2. The isotopic composition is calculated as a function of irradiation time, assembly average exposure, and subsequent decay using the LEOPARD [1] and CINDER[18] computer programs. Once the isotopic compositions of the fuel assemblies are known, the subsequent criticality calculations for the spent fuel racks in Region 2 are performed in a manner that is analogous to the calculations for Region 1.

The accuracy of the exposure dependent isotopic concentrations calculated with the LEOPARD program is demonstrated in Figure 4.2-1 through Figure 4.2-11. Figures 4.2-1 through 4.2-8 show comparisons of LEOPARD calculated data with measured data from a UO_2 fuel assembly irradiated in the Yankee-Rowe reactor while Figures 4.2-9 through 4.2-11 show corresponding data for a mixed oxide (PuO_2-UO_2) fuel assembly irradiated in the SAXTON reactor.

Except for the data labeled PLG calculation, the data and curves on Figures 4.2-1 through 4.2-8 and Figures 4.2-9 through 4.2-11 are taken directly from References 19 and 20, respectively. In all cases, the accuracy of the calculations labeled PLG is within the uncertainty in the measured data.

The accuracy of reactivity calculations for irradiated fuel can be demonstrated in part by the analysis of critical arrays of mixed oxide fuel rods which contain high concentrations of the plutonium isotopes.

Tables 4.2-1 and 4.2-2 show results of criticality analyses for the SAXTON (8) and ESADA(9) sets of experiments which cover a wide range of water-to-oxide volume ratios. A summary of these data is shown in Table 4.2-3. For the mixed oxide criticals, the calculated mean k_{eff} is 0.9969 with a standard deviation about this value of 0.0066 Δk.

The other major uncertainty in the calculations for Region 2 is associated with the calculated reduction in fuel assembly reactivity associated with the depletion of the heavy metals and the accumulation of fission products as a function of fuel assembly exposure. As an example, consider a 4.0 w/o (initial enrichment) Fort Calhoun fuel assembly at 36,000 MWD/MT. The total reactivity loss from the fresh unirradiated case is 0.238 $\Delta k/k$, of which approximately 50% can be attributed to the build-up of fission products. Calculations of reactor reactivity lifetimes using the same analytical methods as used in this analysis demonstrate an accuracy of better than ±5% Therefore, the resulting uncertainty in the calculated fuel assembly k_∞ associated with fuel depletion would be conservatively estimated at 0.0119 $\Delta k/k$ (= .05 x .238 $\Delta k/k$). The corresponding uncertainty in the calculated Region 2 multiplication factor is 0.0107 Δk on a base case Region 2 k_∞ of 0.8960.

In order to provide further assurance of the conservative nature of these calculations, the decay of all fission products following discharge of the fuel assembly was taken into account. This was accomplished with the aid of the CINDER(18) code which treats a total of 186 nuclides in 84 linear chains. The fission product inventory for each fuel assembly was decayed for thirty years following its removal from the reactor core, and the time point of minimum fission product absorption within the thirty year period was used as the basis for determining the fission product

macroscopic absorption cross sections for that particular fuel assembly at that specific exposure. The minimum occurs at approximately 100 days into the decay and from then on continues to increase as illustrated in Figure 4.2-12. Reduction in the fission product inventory due to leakage or escape to the plenum has been found to be negligible. [21]

4.3 CALCULATIONAL APPROACH

4.3.1 Calculational Approach Region 1

The calculational approach used the basic cell to calculate the reactivity of an array of uniform spent fuel racks and to account for any reactivity perturbations of the actual spent fuel rack array versus the assumed infinite array. The effects of manufacturing tolerances, as well as thermal uncertanties, including fuel and water temperature and density variations, are also treated as perturbations on the calculated reactivity of the basic cell.

The adequacy of the calculational mesh selected for this type of cell calculation has been verified by comparison with the results of an identical geometry which used a finer calculational mesh (two times the number of mesh intervals in each direction). The finer calculational mesh resulted in a minor change in the value of k_∞, with an observed increase of +0.0004 Δk.

The fuel assemblies used for this analysis are characterized by the Combustion Engineering "14 x 14" design for which data are provided in Table 4.3-1. The fuel assemblies were assumed to be unirradiated with an assembly enrichment of 4.0 weight percent U-235. This is equivalent to a loading of 46.61 gm of U-235 per axial centimeter of fuel assembly. All of the calculations were performed at a uniform pool temperature of 68°F, except for instances where the reactivity effects of pool temperature were taken into account as a perturbation on the basic cell calculations.

4.3.2 Calculational Approach Region 2

Reactivity calculations, using the LEOPARD and PDQ-7 models described previously for the Region 1 analysis, were performed for Region 2. The geometry used for these calculations is the same as that used for the Region 1 analysis as illustrated in Figure 4.1-1 except that Boraflex® is not present. The specific geometry and dimensions of this basic cell are shown in Figure 4.3-1.

Implementation of this specific design for Region 2 of the pool was based on the selection of a base k_∞ which does not include tolerances or uncertainties. For example, the use of a base k_∞ value of 0.91 assures that the final k_∞ value including all uncertainties and tolerances for Region 2 will be less than 0.95. Figure 4.3-2 provides the Fort Calhoun fuel assembly k_∞ at 68°F, with a minimum fission product inventory, as a function of exposure for several selected initial fuel assembly enrichments. Figure 4.3-3 shows the Region 2 spent fuel storage rack k_∞ as a function of fuel assembly k_∞ based on several selected initial enrichments over a range of fuel assembly exposures.

Finally, the limiting curves presented in Figure 4.3-4, are constructed for a range of spent fuel storage rack base k_∞ values based on Figures 4.3-2 and 4.3-3. These curves show the limiting exposure (i.e., the minimum acceptable exposure) which must be achieved in any Fort Calhoun fuel assembly of the indicated initial enrichment to qualify that fuel assembly for transfer from Region 1 to Region 2 of the pool. By observing these limits on fuel assembly exposure, a safe subcritical margin is maintained in Region 2. Thus, by selection of appropriate design criteria, calculation of appropriate limits on minimum fuel assembly exposure, and implementation of appropriate controls on fuel assembly movements, large numbers of irradiated fuel assemblies may be safely stored in Region 2 without undue risk to the health of both the public and plant personnel.

4.4 MANUFACTURING TOLERANCES AND THERMAL CONSIDERATIONS

4.4.1 Manufacturing Tolerances and Thermal Considerations Region 1

The worst case in terms of manufacturing tolerances, from a reactivity perspective, is represented by the maximum fuel box dimensions together with the minimum water channel width between fuel boxes. This minimum pitch combination of dimensions (9.90 inches) results in the maximum thickness of water surrounding the fuel assembly, which maximizes the fuel neutron absorption fraction; and also results in the minimum amount of water between Boraflex® neutron absorber sheets, which minimizes the thermalization of neutrons in the narrow flux trap between fuel boxes and thus minimizes the absorption fraction in the Boraflex®. The dimensions provided in Figure 4.1-1 describe this situation for the Boraflex® at a maximum thickness of 0.095 inches. The k_∞ of this minimum pitch spent fuel storage rack cell at 68°F for a fuel assembly enrichment of 4.0 weight percent U-235 is 0.9349. Figure 4.4-1 presents the spent fuel storage rack reactivity as a function of fuel assembly enrichment.

For the average rack cell, the pitch will be 9.935 inches, and the k_∞ for this cell is 0.9258. Thus, the perturbation in k_∞ due to tolerances on fuel box cell dimensions is +0.0091 Δk.

If the minimum thickness of the stainless steel box wall itself is considered, which is 0.086 inches compared to the nominal thickness of 0.09 inches, the resulting perturbation is +0.0009 Δk. The wrapper has a nominal thickness of .024 inches ± .002 inches tolerance, the resulting pertubation due to the tolerance is +.0002 Δk.

The reactivity of the basic cell as a function of B^{10} loading in the Boraflex® is shown in Figure 4.4-2. The B^{10} loading which was used for the criticality analysis was the minimum loading incorporated into the design. This corresponds to a B^{10} loading of 0.020 grams per square centimeter of cross sectional area with a nominal thickness of 0.085 inches ± inches 0.01 inches. Increasing the thickness to the maximum value of 0.095 inches, while maintaining the minimum loading of .020 gm B^{10}/cm^2 results in the most limiting k_∞. These assumptions were used in performing the basic cell reactivity calculation.

With regard to fuel position uncertainties within the fuel boxes, calculations confirm that fuel assemblies located in their normal central position in the fuel box are in the most reactive position of the fuel box. The perturbation on the basic spent fuel rack cell is therefore considered to be 0.0 Δk for the fuel positioning uncertainty.

The reactivity of the basic cell as a function of temperature is shown in Figure 4.4-3. With a maximum pool temperature of 200°F, the k_∞ is less than the value of k_∞ at 68°F. This is expected for a design which incorporates a heavy loading of B^{10} as a neutron poison and indicates that the lower temperature conditions produce a higher spent fuel k_∞. Since the reference case was based on a temperature of 68°F, which is clearly conservative, including a reactivity effect for temperature is not necessary .

The sensitivity of the spent fuel rack multiplication factor to variations in the water density throughout the pool is illustrated in Figure 4.4-4. The heavy B^{10} loading produces the most reactive condition at full water density. The large flow area provided by this rack design essentially makes it impossible to trap steam or air in the water boxes when the fuel boxes are filled with water.

The reactivity of the spent fuel storage rack was evaluated for the effect of manufacturing tolerances on UO_2 density. The reference cell is based on a fuel design value of 94% theoretical density. The worst case examined was UO_2 at a density of 96% theoretical density. The resulting perturbation to the basic cell was determined to be .0023 Δk due to the increase in pellet density from 94% to 96%. However, the maximum pellet density to be allowed in the fuel asemblies is 95.5% and for this density the corresponding pertubation is .0017Δk.

There are two axial effects which result in perturbations of the basic cell Δk which assumes an infinite and uniform height of fuel and poison. The poison material is 120 inches in height which leaves 4 inches at the top and bottom of the active fuel without poison material. Omission of this poison material increases the calculated k_∞ by 0.0012 Δk. Since the active fuel height is only 128 inches there is a small but significant leakage of neutrons from the top and bottom of the fuel assemblies. This perturbation in k_∞ is calculated to be -0.0027 Δk. Therefore, the net perturbation in k_∞ due to axial affects is:

$$+0.0012 - 0.0027 = -0.0015$$

A summary of the perturbations to the basic cell reactivity calculations is shown in Table 4.4-1. As shown in Table 4.4-1, the conservatively calculated reactivity of the spent fuel rack fully loaded with unirradiated bundles, 4.0 weight percent U-235 and no burnable poison is 0.9415, at a pool temperature of 68°F for the most pessimistic manufacturing conditions and including calculational uncertainties. A more realistic estimate of the maximum reactivity which would be achieved in the racks is also shown in Table 4.4-1. With the minimum allowable concentration of soluble boron in the pool water (i.e., 1700 ppm), the maximum k_∞ is 0.7411.

4.4.2 Manufacturing and Thermal Considerations Region 2

The reactivity perturbations associated with manufacturing and thermal tolerances and uncertainties as noted for Region 1 were re-evaluated for depleted fuel for applicability in the Region 2 configuration. This re-evaluation assumed Fort Calhoun fuel characterized by a 4.0 w/o U-235 initial enrichment at 36,000 MWD/MT exposure. This fuel assembly and exposure were selected as being typical of the irradiated fuel being considered for storage.

Specific calculations were performed on the irradiated fuel for variations in all relevant physical dimensions, temperature, and water density. Table 4.4-2 provides a summary of the reactivity perturbations to the Region 2 spent fuel storage racks. Detailed results of these calculations for Region 2 are presented in Figure 4.4-5 and 4.4-6 for the effects of temperature and water density, respectively.

As shown in Table 4.4-2, the conservatively calculated reactivity of the spent fuel rack fully loaded with irradiated bundles, 4.0 w/o U-235 is 0.9459, at a pool temperature of 68°F for the most pessimistic manufacturing conditions and calculational uncertainties. With the minimum allowable concentration of soluble boron in the pool water (i.e., 1700 ppm), the maximum k_∞ is 0.6380.

4.5 DESIGN CONSERVATISMS

The High Density Rack design concept for Region 1 is based on currently accepted conservative criteria which allow for the safe storage of a number of fresh unirradiated fuel assemblies (including a full core unloading if that should prove necessary). The penalty in achievable spent fuel storage density associated with this conservative design assumption was relatively small under the circumstances originally anticipated and easily accommodated by a conservative spent fuel rack design. The potential penalty associated with this conservative design basis is no longer small when long term on-site storage of spent fuel is a possibility.

It is not conceivable that more than one full core load of fresh unirradiated fuel assemblies could be stored in the spent fuel storage pool. Therefore, it is unnecessary and wasteful to base the entire spent fuel storage rack design on the assumption of fresh unirradiated fuel of the highest initial enrichment.

While the High Density Rack concept reduces some of the design conservatisms inherent in the earlier spent fuel storage concepts, the design and analyses for the High Density Rack as immplemented in Region 2 are still conservative in nature.

The use of assembly average exposure is one example of this conservative approach. Axially, more than 80% of the fuel assembly will normally have reached exposures greater than the average and this will occur in the central, higher worth region of the assembly. The lower exposure regions would normally account for less than 20% of the fuel assembly length distributed at the ends of the fuel assembly active length in lower worth regions. The result is a neutronically higher exposure assembly than represented by the simple assembly average exposure. The use of the simple assembly average exposure can result in an over-estimate of the fuel assembly k_{eff} by about $+.015\ \Delta k/k$.

4.6 ACCIDENT ANALYSIS REGION 1 and 2

As reported in Tables 4.4-1 and 4.4-2 the largest k_∞ of the basic cell with 1700 ppm boron in the water is 0.7411 and therefore an extremely large reactivity perturbation (i.e., greater than .2589 Δk) would be required to produce a criticality accident.

The fuel racks are designed to prevent a dropped fuel bundle from penetrating and occupying a position other than a normal fuel storage location. The only potential significant positive reactivity effect of a dropped fuel bundle is the reduction in axial neutron leakage that could occur if the dropped bundle came to rest in a horizontal position on top of the racks.

However, the top of the active fuel stored in the racks is about 27 inches below the top of the storage boxes. This thickness of water is greater than that required to neutronically decouple the dropped fuel assembly from the assemblies stored in the rack, and therefore the maximum possible k_{eff} for Region 1 remains unchanged for this assumed accident (i.e., a value of .9415 in unborated water and .7411 with 1700 ppm boron in the pool) Similarly, for Region 2 the maximum possible k_{eff} remains unchanged for this assumed accident (i.e., a value of 0.9459 in unborated water and 0.6380 with 1700 ppm boron in the pool).

The reactivity effect of a fresh fuel assembly located adjacent to a fully loaded spent fuel storage rack has been evaluated for all postulated locations other than normal fuel storage locations. In areas where there is sufficient space to physically allow a fresh fuel assembly to be placed immediately adjacent to a rack storage box, stand offs have been provided; so that, in all cases, the spent fuel storage rack design assures that the multiplication factor will be less than 0.95.

The lattice of the fuel assemblies results in an undermoderated configuration so that any crushing or compaction of the fuel assemblies would tend to reduce the multiplication factor of the spent fuel pool. Therefore, the dropping of heavy objects into the fuel pool or deformations from the effects of earthquakes or tornadoes will not produce a criticality accident. The relative positions of the fuel racks are maintained by perimeter bars on the rack therefore a significant reactivity perturbation due to rack movement is precluded. The criticality safety analysis is based on the minimum spacing between adjacent racks, and any rack movement which would increase this spacing would tend to reduce the k_∞ of the racks.

Loss of all cooling systems would result in a pool water temperature increase and under the worst possible conditions, boil off of the pool water. Both higher temperature and lower density water result in negative reactivity pertubations for the most limiting conditions for the criticality analysis and therefore loss of all cooling systems would not produce a criticality accident.

Because of the well founded, conservative technique used for determination of the infinite multiplication factor, there is more than reasonable assurance that this spent fuel rack design will not cause undue risk to the public health and safety resulting from criticality considerations.

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TABLE 4.0-1

GENERAL CRITICALITY DESIGN CRITERIA

Region 1 Criteria	Region 2 Criteria
1. Fresh, unirradiated fuel inventory is assumed.	1. Actual irradiated fuel and fission product inventory is assumed.
2. $k_{eff} < 0.95$	2. Same.
3. Credit may be taken for presence of borated water for abnormal (accident) conditions.	3. Same.
4. Fuel location recorded once prior to storage in Region 1.	4. A check is required for each fuel assembly prior to transfer from Region 1 to Region 2.

TABLE 4.1-1
SUMMARY OF LEOPARD RESULTS FOR MEASURED CRITICALS

Case** No.	Reference No.	Enrichment (atom %)	H_2O/U Volume	Fuel Density (g/cm^3)	Pellet Diameter (cm)	Clad Diameter (cm)	Clad Thickness (cm)	Lattice Pitch (cm)	Critical Buckling (m^{-2})	Calculated k_{eff}
1	11	2.734	2.18	10.18	0.7620	0.8594	0.04085	1.0287	40.75	1.0015
2	11	2.734	2.93	10.18	0.7620	0.8594	0.04085	1.1049	53.23	1.0052
3	11	2.734	3.80	10.18	0.7620	0.8594	0.04085	1.1938	63.28	1.0043
4	12	2.734	7.02	10.18	0.7620	0.8594	0.04085	1.4554	65.64	1.0098
5	12	2.734	8.49	10.18	0.7620	0.8594	0.04085	1.5621	60.07	1.0118
6	12	2.734	10.43	10.18	0.7620	0.8594	0.04085	1.6891	52.92	1.0072
7	13	2.734	2.50	10.18	0.7620	0.8594	0.04085	1.0617	47.5	1.0008
8	13	2.734	4.51	10.18	0.7620	0.8594	0.04085	1.2522	68.8	0.9937
9	13	3.745	2.50	10.37	0.7544	0.8600	0.0406	1.0617	68.3	1.0010
10	13	3.745	4.51	10.37	0.7544	0.8600	0.0406	1.2522	95.1	1.0025
11	14	3.745	4.51	10.37	0.7544	0.8600	0.0406	1.2522	95.68	1.0009
12	15	4.099	2.55	9.46	1.1278	1.2090	0.0406	1.5113	88.0	0.9839
13	15	4.069	2.14	9.46	1.1278	1.2090	0.0406	1.450	79.0	0.9830
14	16	4.069	2.59	9.45	1.1268	1.2701	0.07163	1.555	69.25	0.9999
15	16	4.069	3.53	9.45	1.1268	1.2701	0.07163	1.684	85.52	0.9958
16	16	4.069	8.02	9.45	1.1268	1.2701	0.07163	2.198	92.84	1.0040
17	16	4.069	9.90	9.45	1.1268	1.2701	0.07163	2.381	91.79	0.9882

TABLE 4.1-1 (Continued)

SUMMARY OF LEOPARD RESULTS FOR MEASURED CRITICALS

Case** No.	Reference No.	Enrichment (atom %)	H_2O/U Volume	Fuel Density (g/cm^3)	Pellet Diameter (cm)	Clad Diameter (cm)	Clad Thickness (cm)	Lattice Pitch (cm)	Critical Buckling (m^{-2})	Calculated k_{eff}
18	16	3.037	2.64	9.28	1.1268	1.2701	0.07163	1.555	50.75	0.9946
19	16	3.037	8.10	9.28	1.1268	1.2701	0.07163	2.198	68.81	0.9809
20	8	0.714*	1.68	9.52	0.8570	0.9931	0.0592	1.3208	108.8	0.9912
21	8	0.714*	2.17	9.52	0.8570	0.9931	0.0592	1.4224	121.5	1.0029
22	8	0.714*	4.70	9.52	0.8570	0.9931	0.0592	1.8669	159.6	0.9944
23	8	0.714*	10.76	9.52	0.8570	0.9931	0.0592	2.6416	128.4	1.0008
24	9	0.729*	1.11	9.35	1.2827	1.4427	0.0800	1.7526	89.1	0.9902
25	9	0.729*	3.49	9.35	1.2827	1.4427	0.0800	2.4785	104.72	1.0055
26	9	0.729*	3.49	9.35	1.2827	1.4427	0.0800	2.4785	79.5	0.9948
27	9	0.729*	1.54	9.35	1.2827	1.4427	0.0800	1.9050	90.0	0.9878

* These are PuO_2 in Natural UO_2

** Case 1 through 19 are with stainless steel clad, Cases 20 through 27 are zircalloy

TABLE 4.1-2

WESTINGHOUSE UO_2 Zr-4 CLAD CYLINDRICAL CORE CRITICAL EXPERIMENTS

EXPERIMENT	PITCH (in)	BORON CONCENTRATION (ppm)	MATERIAL BUCKLING (FOR LEOPARD) (cm^{-2})	CRITICAL NO. OF PINS (cm)	RADIUS OF FUEL REGION	k_{eff} (LEOPARD/PDQ-7)
1	0.600	0	.008793	489.4	19.021	0.9912
2	0.690	0	.009725	317.0	17.605	0.9941
3	0.848	0	.008637	251.6	19.276	0.9927
4	0.976	0	.006458	293.0	23.935	0.9935
5	0.600	306.	.007177	659.9	22.088	0.9927
6	0.600	536.4	.006244	807.2	24.429	0.9937
7	0.600	727.7	.005572	950.2	26.504	0.9940
8	0.600	104.	.008165	546.3	20.097	0.9919
9	0.600	218.	.007599	607.1	21.185	0.9917
10	0.600	330.	.007106	669.5	22.248	0.9916
11	0.600	446.	.006661	735.3	23.315	0.9909
12	0.600	657.1	.005809	895.3	25.727	0.9944
13	0.848	104.	.007320	321.0	21.772	0.9938
14	0.848	218.	.006073	420.5	24.919	0.9925

TABLE 4.1-2 (Continued)

WESTINGHOUSE UO_2 Zr-4 CLAD CYLINDRICAL CORE CRITICAL EXPERIMENTS

k_{eff}
(LEOPARD/PDQ-7)

0.9928 Mean
0.0012 Std Deviation

Fuel Region Data

Enrichment	= 2.719 w/o U-235
Fuel Density	= 10.41 g/cm^3
Pellet Radius	= 0.20 in
Clad IR	= 0.2027 in
Clad OR	= 0.23415 in

(b) Thickness of water reflector is that required to attain total radius of 50 cm for model.

(c) B_z^2 (PDQ-7) = .000527 cm^{-2}

TABLE 4.1-3

BATTELLE FIXED NEUTRON POISON CRITICALS (17)

Case	Length Times Width*	No. of Assemblies In Array	Absorber Type	Thickness	Distance To Fuel Cluster	Critical Separation of Clusters	k_{eff} LEOPARD/PDQ
020	20x17	3	Boral	.713 cm	.645 cm	6.34 cm	0.9932
017	22.21x16^x	3	"	"	"	5.22 "	0.9936
002	20x18.88+	1	"	"	2.732 cm	∞	0.9926
028	20x16	3	S.S.	.485 cm	.645 cm	6.88 cm	0.9922
027	20x16	3	S.S.	.302 "	"	7.43 "	0.9919
032	20x17	3	S.S. 1.1w/o B	.298 cm	.645 cm	7.56 cm	0.9921
038	20x17	3	S.S. 1.6w/o B	"	"	7.36 "	0.9928
002B	20x18.075	1	None	-	-	∞	0.9956
015	20x17	3	"	-	-	11.94 cm	0.9932
013	20x16	3	"	-	-	8.42 "	0.9921
022	20x15	3	"	-	-	6.39 "	0.9933
021	20x16	3	"	-	-	4.46 "	0.9946

TABLE 4.1-3 (continued)

BATTELLE FIXED NEUTRON POISON CRITICALS [17]

Statistical Summary:

Series	Number	mean k_{eff}	σ
Boral	3	0.9931	0.0005
S.S.	2	0.9925	0.0005
S.S. (Borated)	2	0.9921	0.0002
Fixed Poison Total	7	0.9926	0.0006
Non-Poison Total	5	0.9938	0.0014
Overall	12	0.9931	0.0011

* This is in units of pitch (Pitch=2.032 cm)

x Center assembly was 20x16 and the other two were elongated at 22.21x16.

+ 20x18.88 was one assembly with a boral sheet on two sides.

Fuel region data: Enrichment=2.35w/o, Pellet radius=0.5588 cm,
Clad OR=.635 cm, Wall thickness=.0762 cm, Pitch=2.032 cm

TABLE 4.2-1

SAXTON PuO_2-UO_2 CRITICAL EXPERIMENTS

(Reference 8)

Expt.	Boron (ppm)	H_2O/UO_2 (Volume)	Pitch (Inches)	k_{eff}	$k_{eff}-1$
1	0	1.68	.520	.9912	-.0088
2	0	2.17	.560	1.0029	+.0029
3	337	2.17	.560	1.0084	+.0084
4	0	4.70	.735	.9944	-.0056
5	0	10.76	1.040	1.0008	+.0008

TABLE 4.2-2

ESADA PuO_2-UO_2 CRITICAL EXPERIMENTS

(Reference 9)

Expt.	Boron (ppm)	Pu-240 (%)	H_2O/UO_2 (Volume)	Pitch (Inches)	k_{eff}	$k_{eff}-1$
1	0	8	1.11	.690	.9902	-.0098
2	0	8	3.49	.9758	1.0055	+.0055
3	526	8	3.49	.9758	.9949	-.0051
4	0	24	3.49	.9758	.9948	-.0052
5	0	8	1.54	.750	.9878	-.0122
6	526	8	1.11	.690	.9945	-.0055

TABLE 4.2-3

SUMMARY OF PREDICTIONS FOR k_{eff}

IN CRITICALITY EXPERIMENTS

Experiment	Cases	k_{eff}
Saxton PuO_2-UO_2	5	0.9995 ± .0068
ESADA PuO_2-UO_2	6	0.9946 ± .0061
All PuO_2-UO_2	11	0.9969 ± .0066

TABLE 4.3-1

FUEL ASSEMBLY TECHNICAL INFORMATION
FOR
FORT CALHOUN NUCLEAR PLANT

Rod Array	14 x 14
Rods Per Assembly	1.76
Rod Pitch, In.	0.580
Overall Dimensions, In.	8.13 x 8.13
Assembly Overall Length, In.	148.83
Active Fuel Height, In.	128.0
Clad Thickness, In.	.026
Fuel Rod O.D., In.	0.440
Pellet Diameter, In.	0.3815
Diametral Gap, In.	0.0065
Pellet Density (% Theoretical)	94
Control Rod Guide Tubes	
Outer Diameter, In.	1.135
Wall Thickness, In.	0.050
Center Guide Tube	
Outer Diameter, In.	1.122
Wall Thickness, In.	0.050

TABLE 4.4-1

SUMMARY OF REACTIVITY BIASES AND UNCERTAINTIES
FOR FORT CALHOUN REGION 1

DESCRIPTION	REACTIVITY EFFECT, Δk	k_∞
Basic cell at 68°F 4.0 w/o U-235, 9.935 inch pitch, (see Figure 4.1-1)		0.9258
Calculation Biases		
LEOPARD/PDQ Model Bias	+ .0072	
Mesh Spacing Effect	+ .0004	
Net Axial Adjustment	- .0015	
	+ .0061	
Basic cell including Biases		0.9319
Tolerances and Uncertainties		
Minimum pitch (i.e., box to box spacing)	.0091	
Wrapper	.0002	
Tolerance in SS box thickness	.0009	
Pellet density (± .015)	.0017	
Pellet diameter (± .0005)	.0003	
Calculational uncertainty (2 σ)	.0024	
Total Uncertainty (statistical)	.0096	
Maximum, including all biases and uncertainties		0.9415
Basic cell at 68°F with 1700 ppm boron		0.7411

TABLE 4.4-2

SUMMARY OF REACTIVITY BIASES AND UNCERTAINTIES
FOR FORT CALHOUN REGION 2

Description	Reactivity Effect, Δk	k_∞
Basic cell at 68°F,4.0 w/o U-235, at 36,000 MWD/MT, 9.935 inch pitch, (see Figure 4.3-1)		0.9094
Calculation Biases		
Most reactive temperature condition over the operating range	+.0191	
Axial Leakage	-.0028	
LEOPARD/PDQ Model Bias	+.0031	
Mesh Spacing Effect	-.0005	
Basic cell including Biases	+.0189	0.9283
Tolerances and Uncertainties		
Depleted fuel assembly reactivity uncertainties	.0107	
Minimum pitch (i.e., box to box spacing)	.0035	
Tolerance in SS box thickness	.0026	
Pellet density (± 015)	.0017	
Pellet diameter (± .0005)	.0003	
Calculational uncertainty (2σ)	.0132	
Total Uncertainty (statistical)	.0176	
Maximum, including all biases and uncertainties		0.9459
Basic cell at 68°F with 1700 ppm boron		0.6380

FIGURE 4.1-1

FORT CALHOUN SPENT FUEL RACK BASIC CELL
FOR
REGION 1 CRITICALITY CALCULATIONS

Cavity Pitch =	9.935	$^{+.085}_{-.035}$
Can Opening =	8.460	± .030
Can Thickness=	.090	± .004
Boraflex Thick=	.085	± .010
Outer Wrapper=	.024	± .002

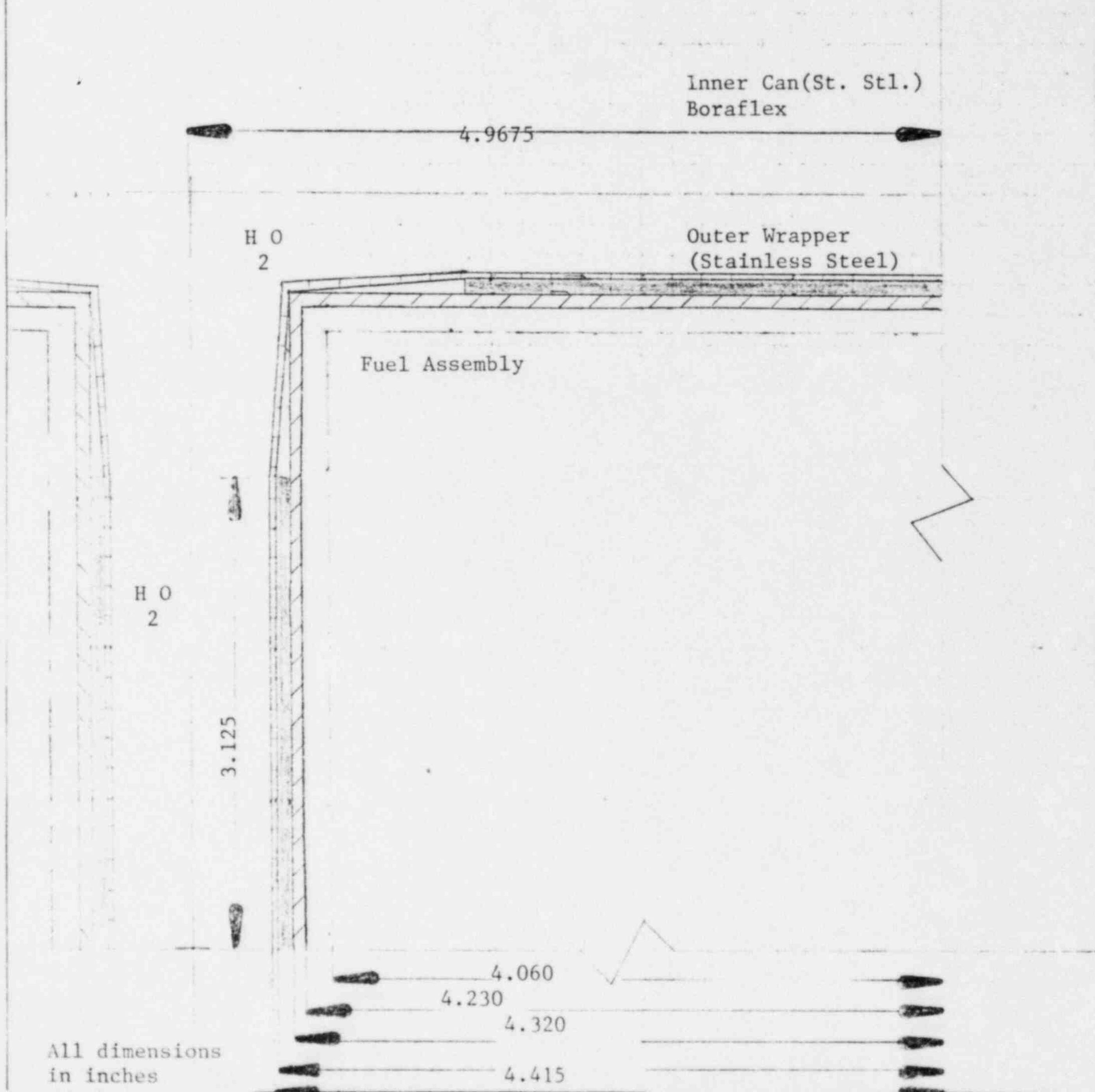
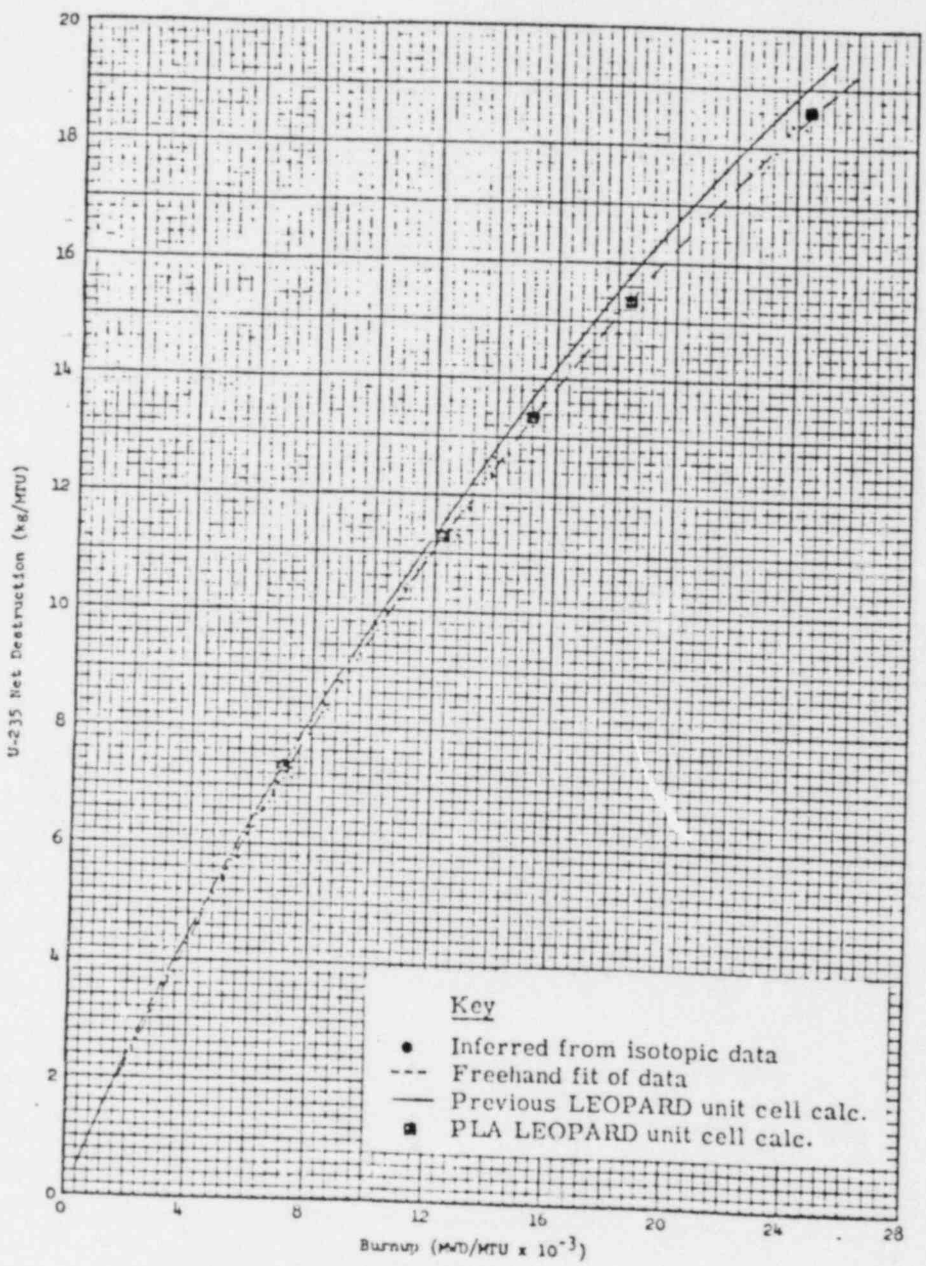


FIGURE 4.2-1

NET DESTRUCTION OF U-235 VERSUS BURNUP
IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM



Rev. 1
Date 12/17/82

FIGURE 4.2-2

SPECIFIC PRODUCTION OF U-236 VERSUS
BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

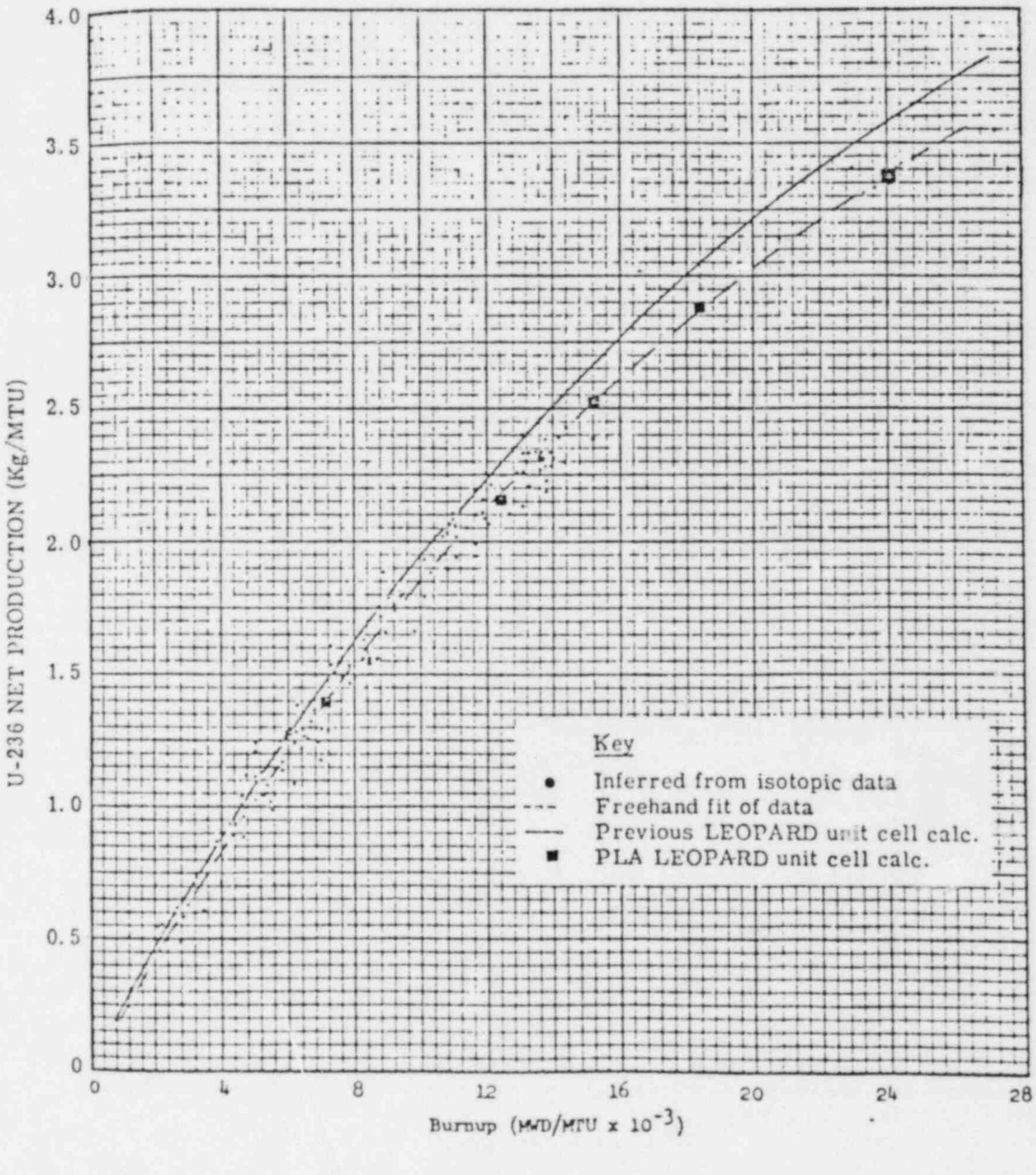
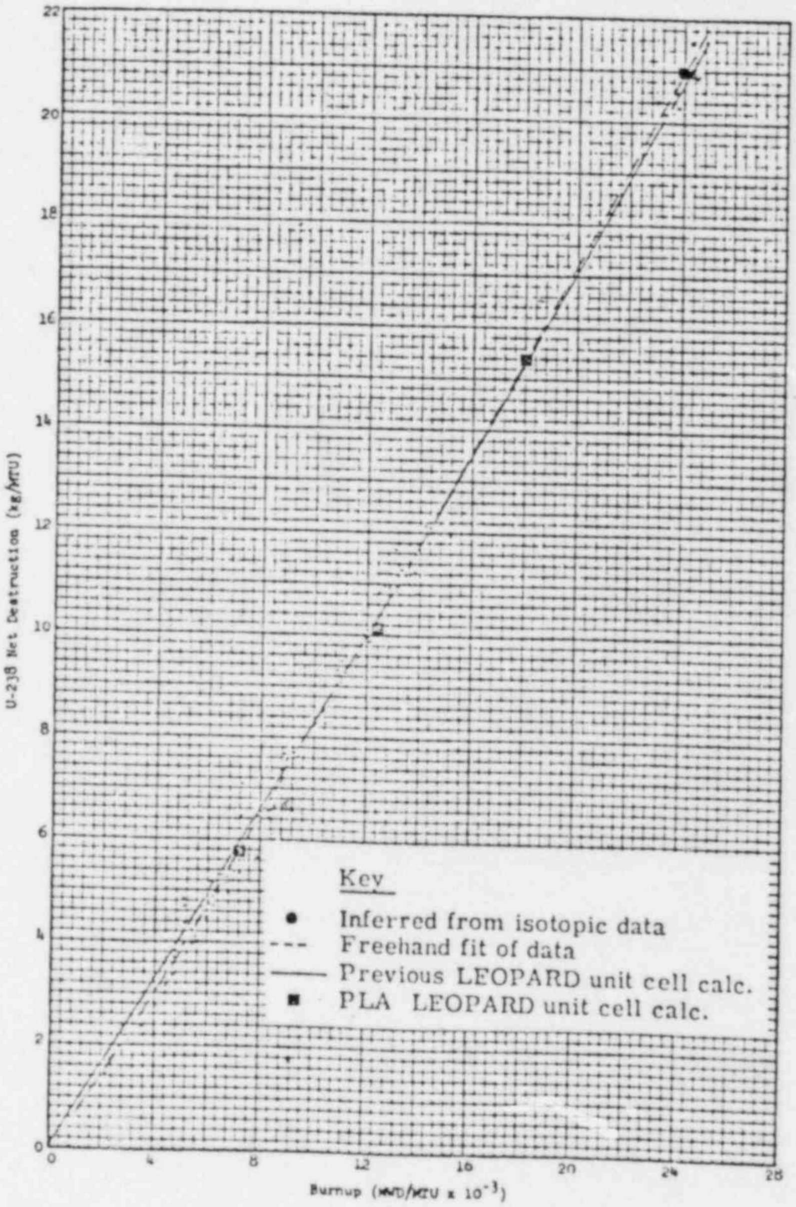


FIGURE 4.2-3

NET DESTRUCTON OF U-238 VERSUS BURNUP
IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM



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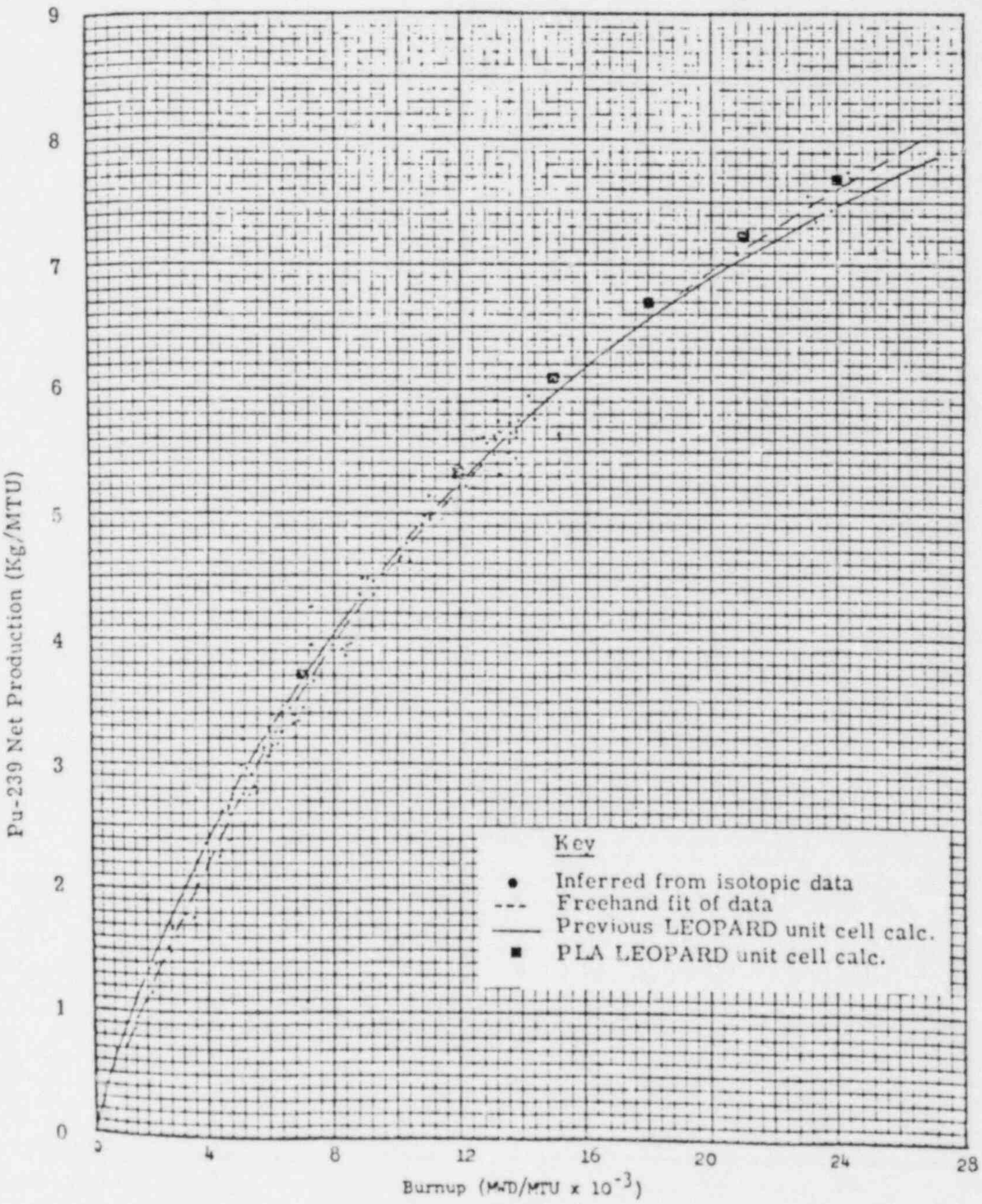
Date 12/17/82

FIGURE 4.2-4

SPECIFIC PRODUCTION OF Pu-239 VERSUS BURNUP
IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM



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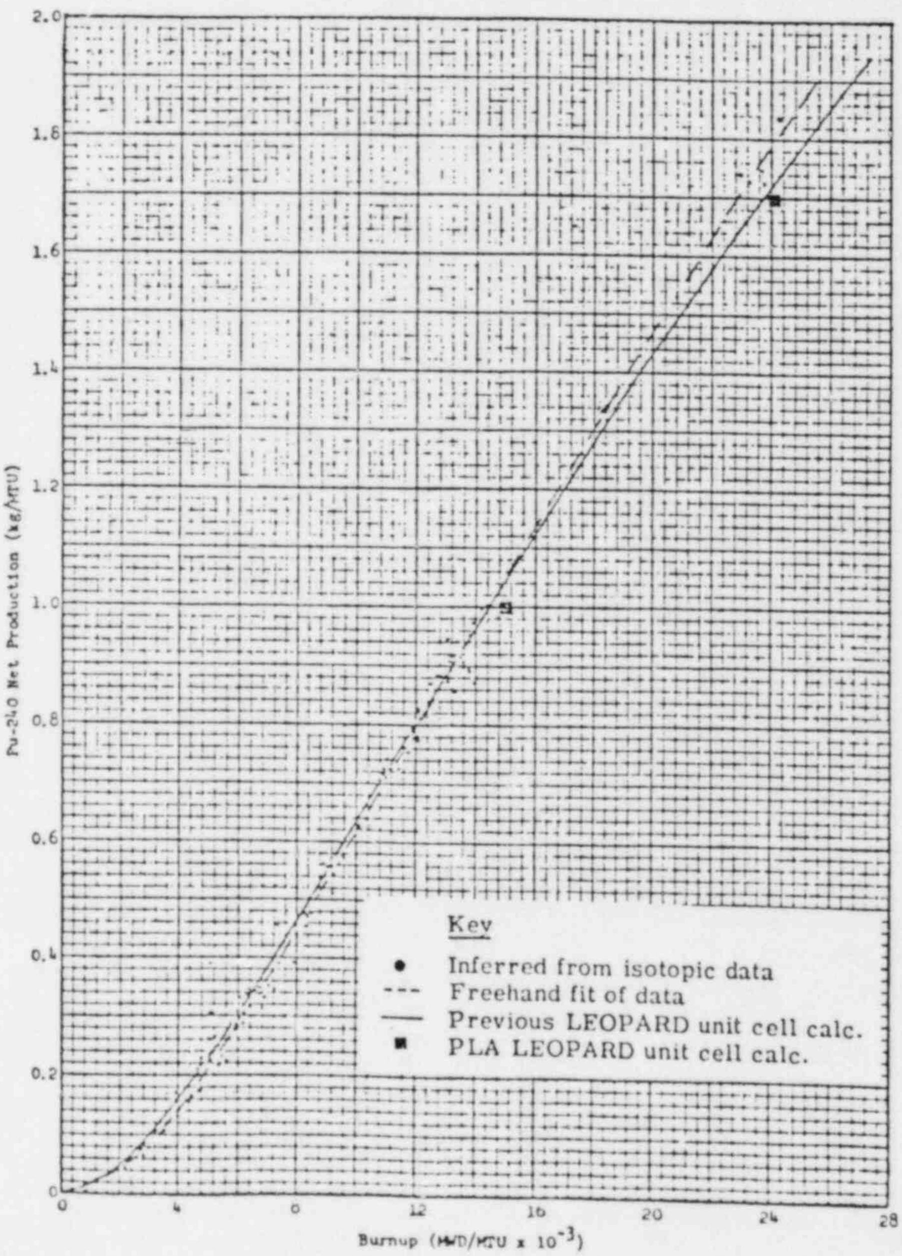
Date 12/17/82

FIGURE 4.2-5

SPECIFIC PRODUCTION OF Pu-240 VERSUS BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM



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Date 12/17/82

FIGURE 4.2-6

SPECIFIC PRODUCTION OF Pu-241 VERSUS BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

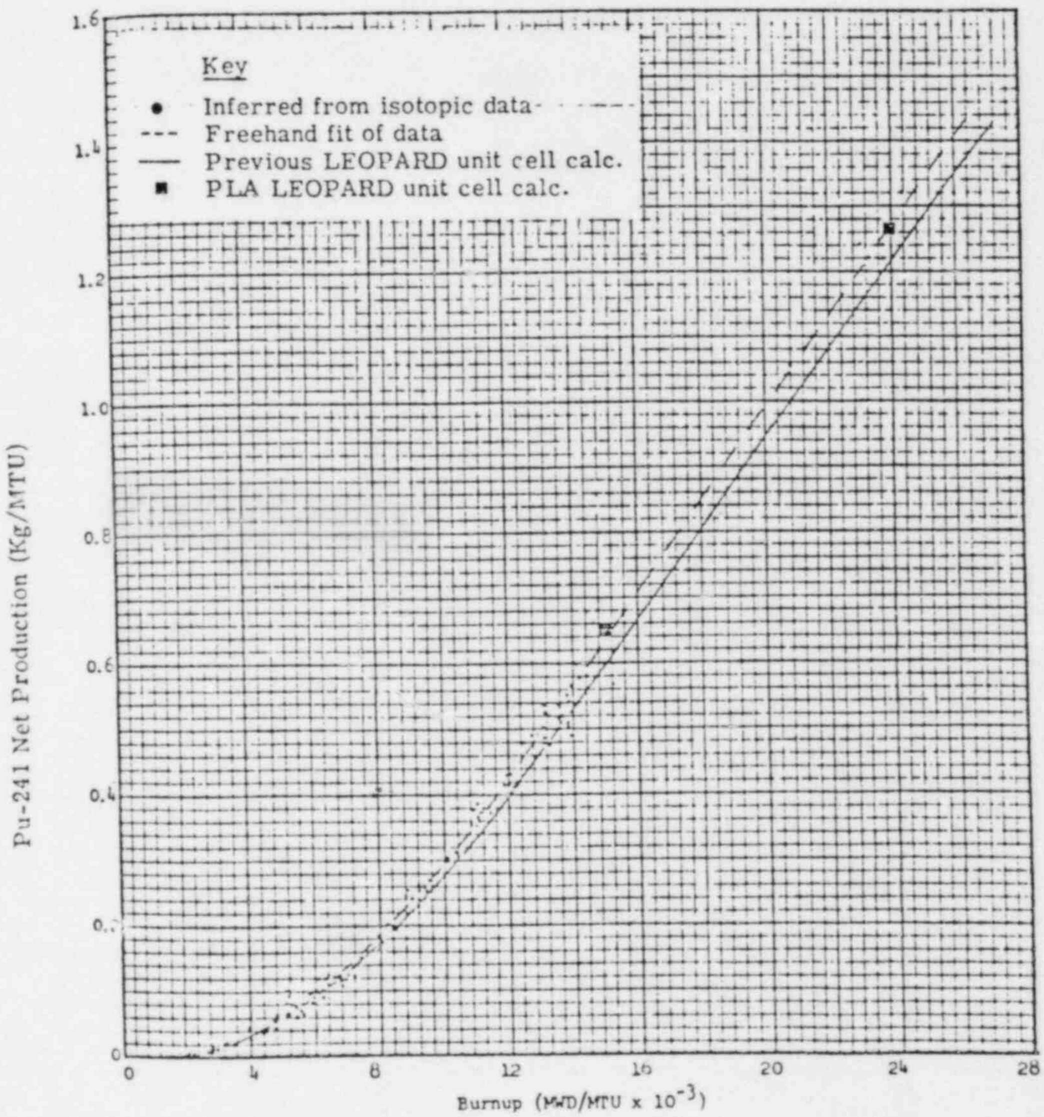


FIGURE 4.2-7

SPECIFIC PRODUCTION OF Pu-242 VERSUS BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

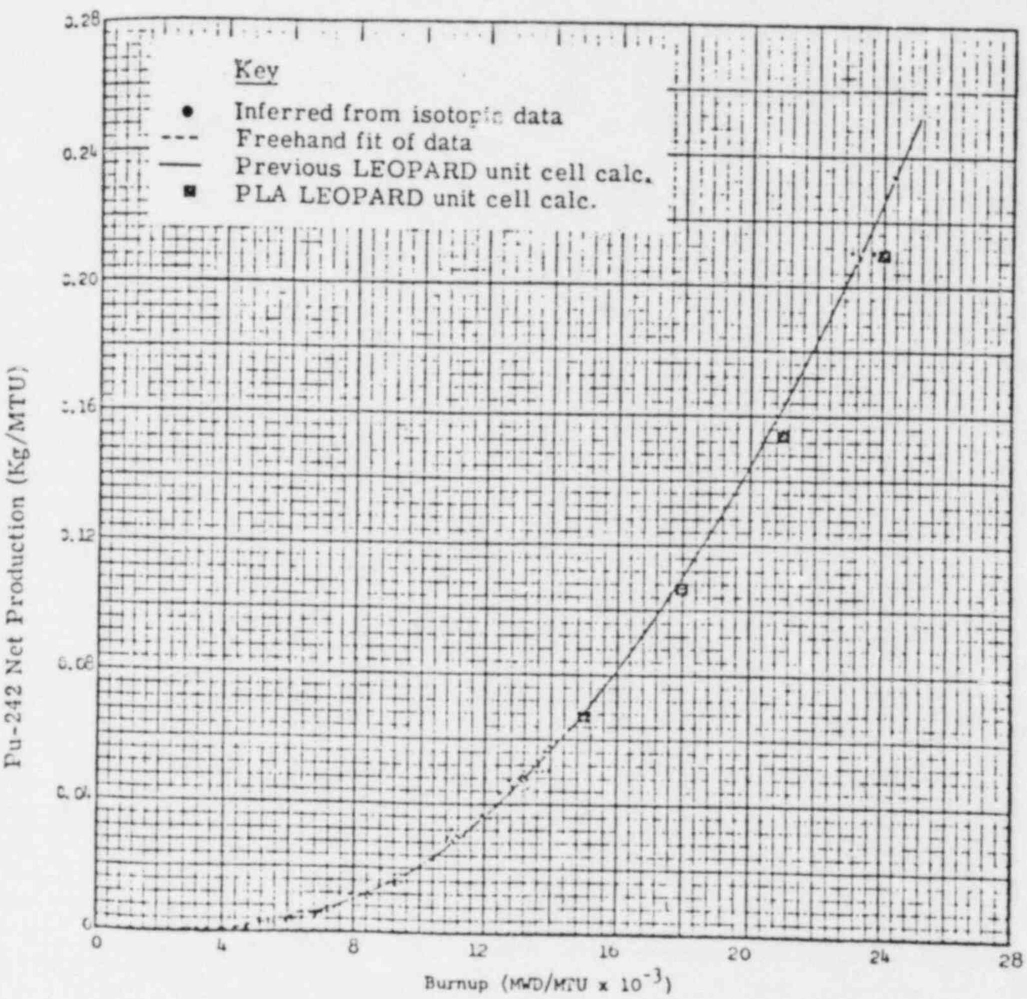


FIGURE 4.2-8

SPECIFIC PRODUCTION OF Pu AND FISSILE Pu VERSUS BURNUP IN THE YANKEE ASYMPTOTIC NEUTRON SPECTRUM

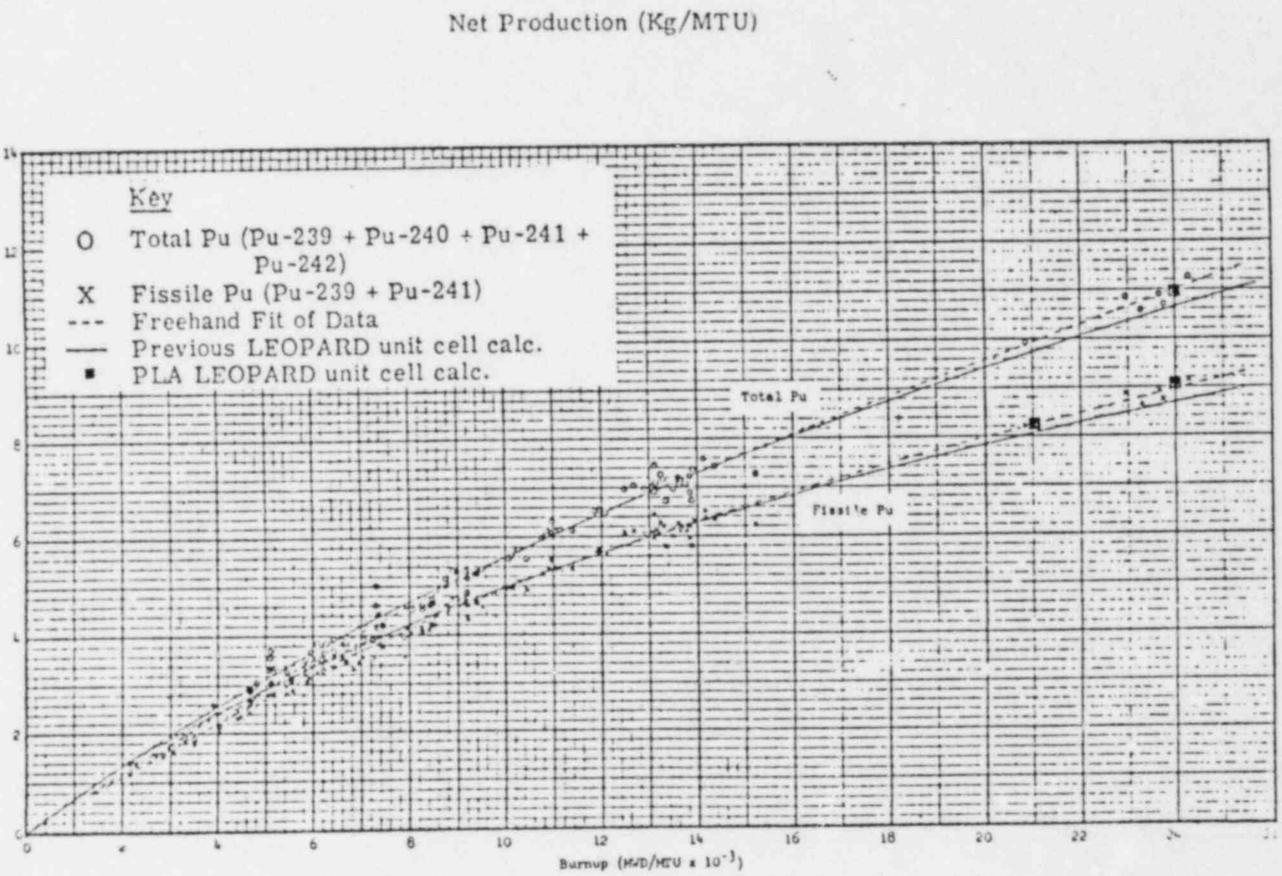


FIGURE 4.2-9

ATOM PERCENT OF TOTAL U VERSUS EXPOSURE

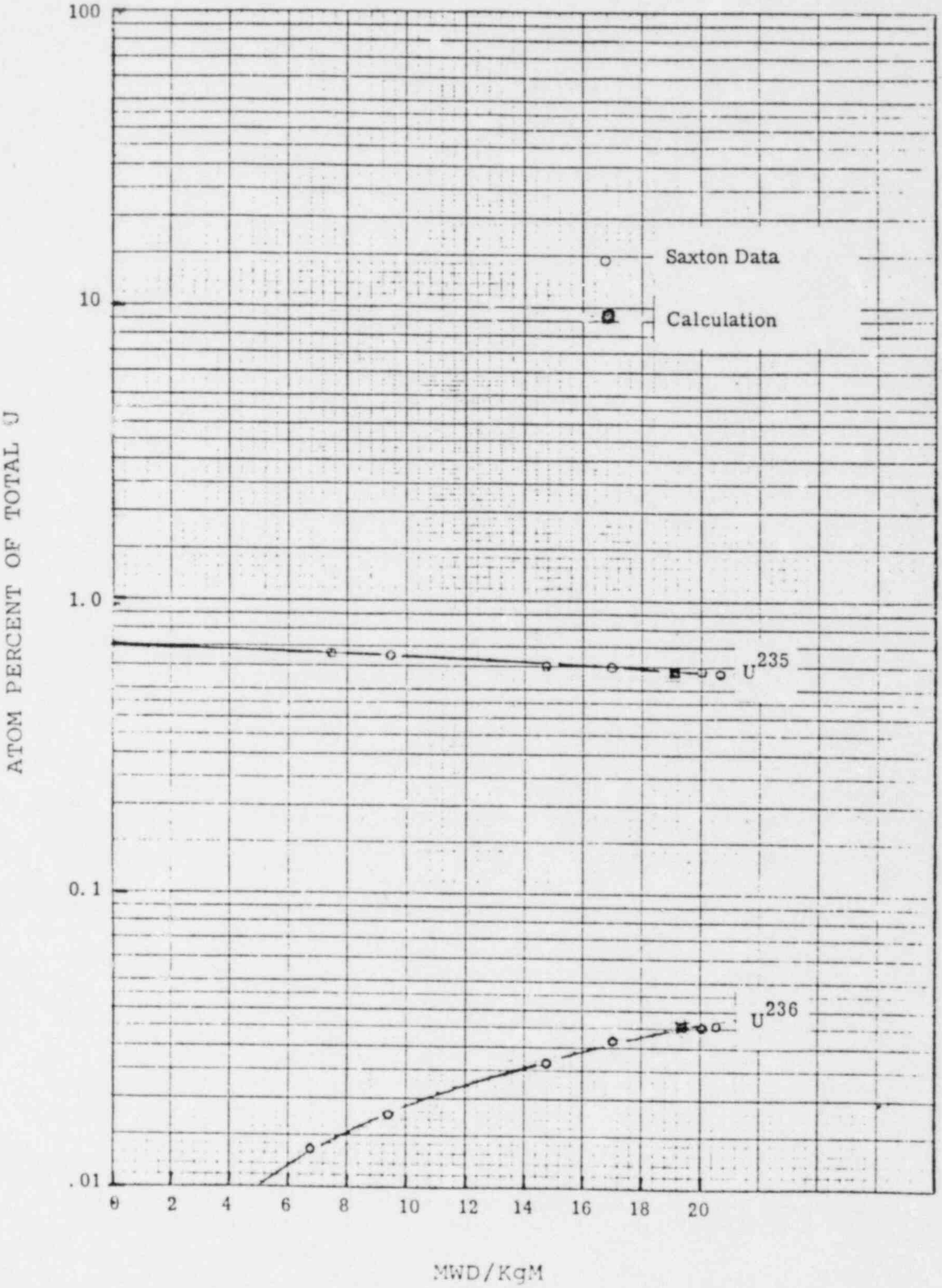


FIGURE 4.2-10

Pu-239/U-238 ATOM RATIO VERSUS EXPOSURE

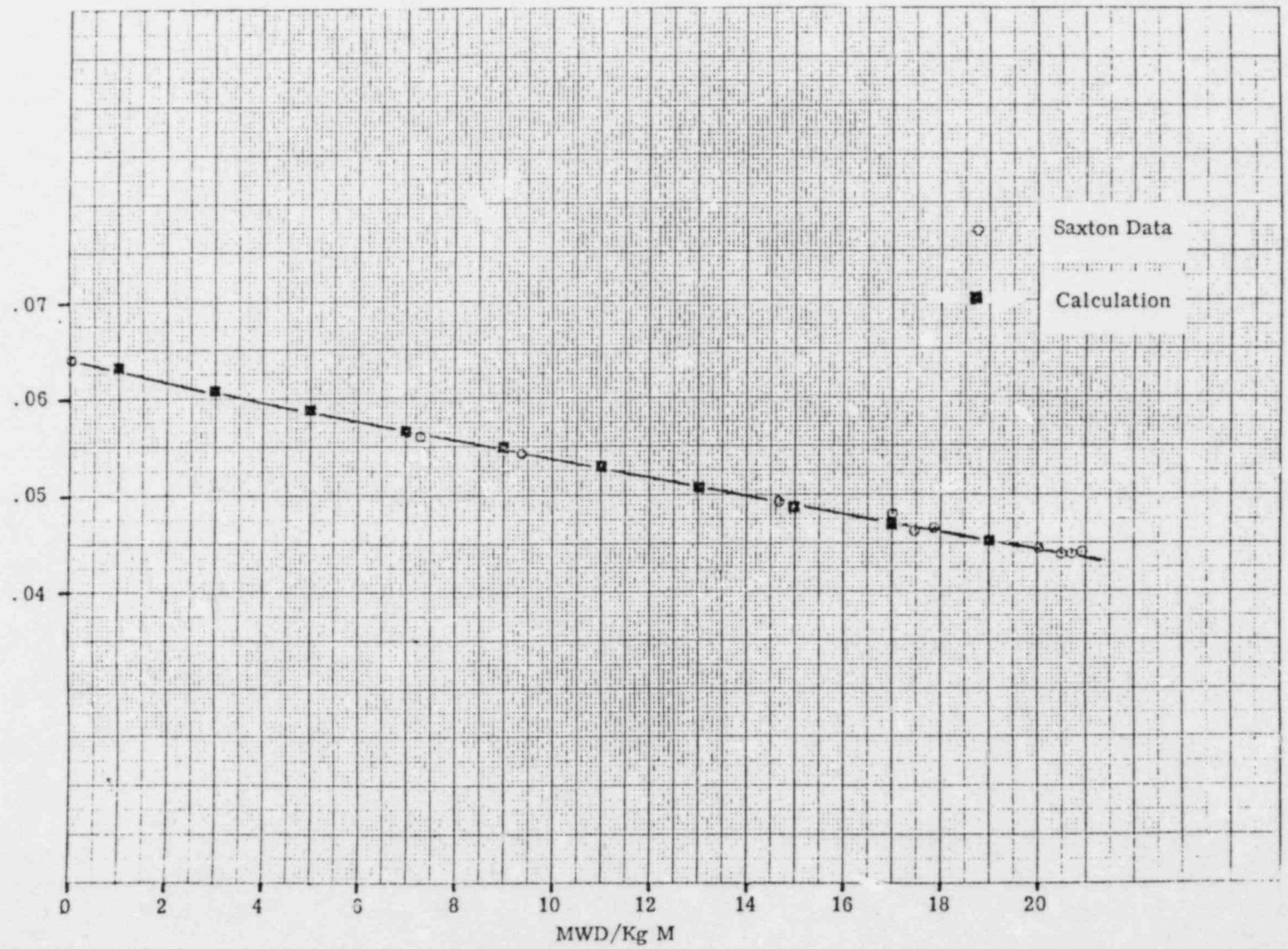


FIGURE 4.2-11

ATOM PERCENT OF TOTAL Pu VERSUS EXPOSURE

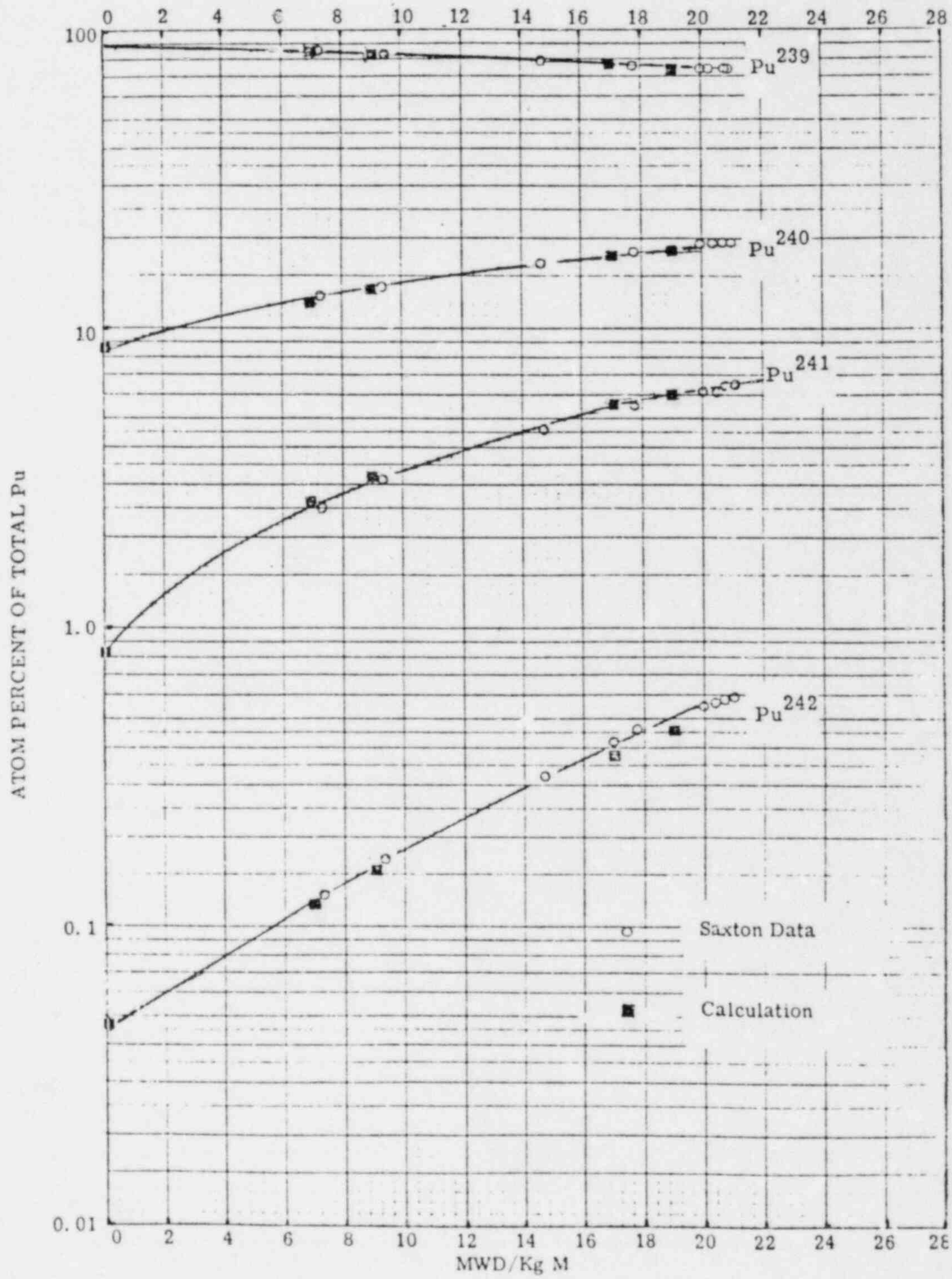


FIGURE 4.2-12

FISSION PRODUCT POISONING - FOR FORT CALHOUN

(Barns per Fission - 2200 m/s vs Decay Time)

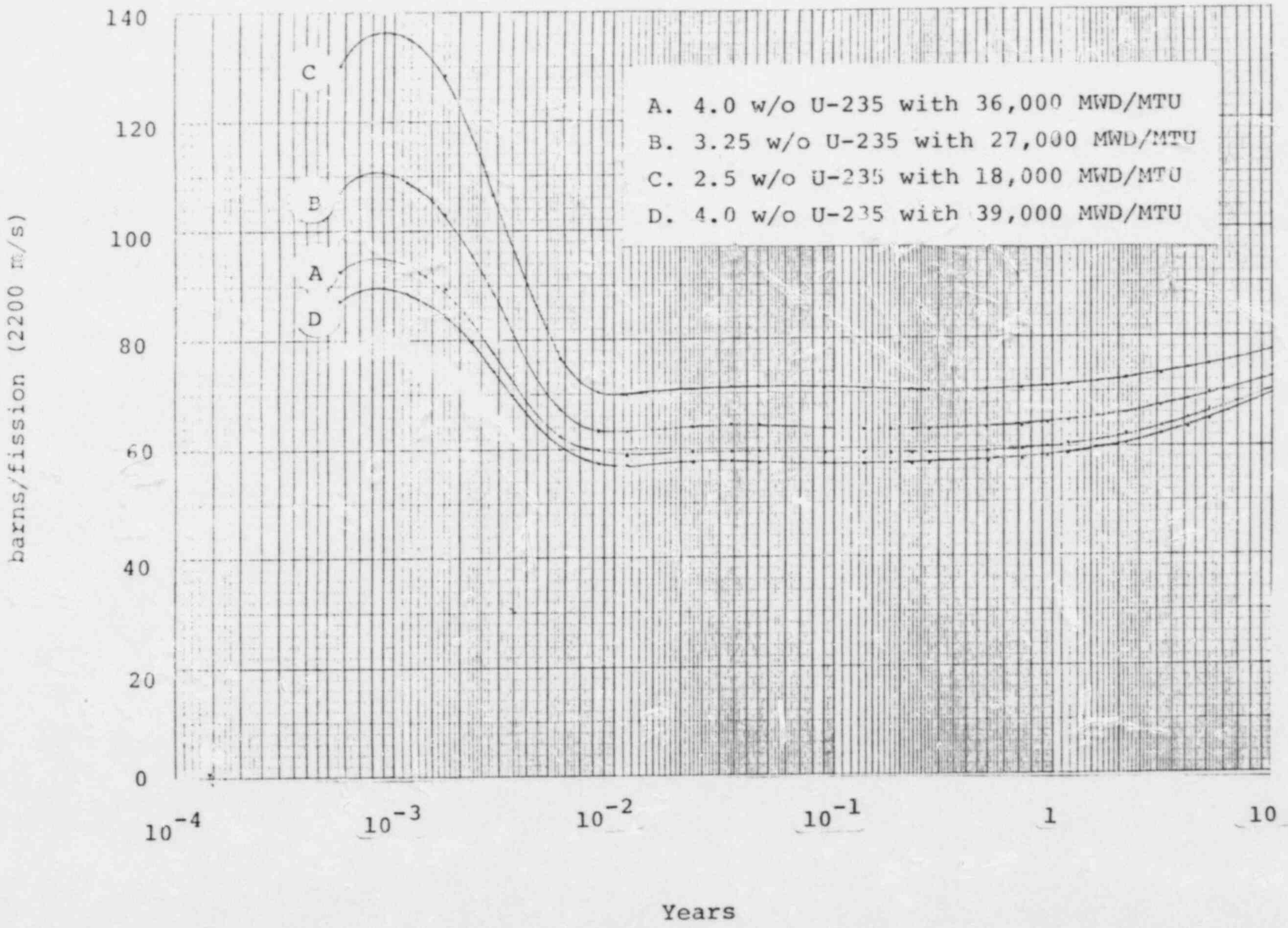
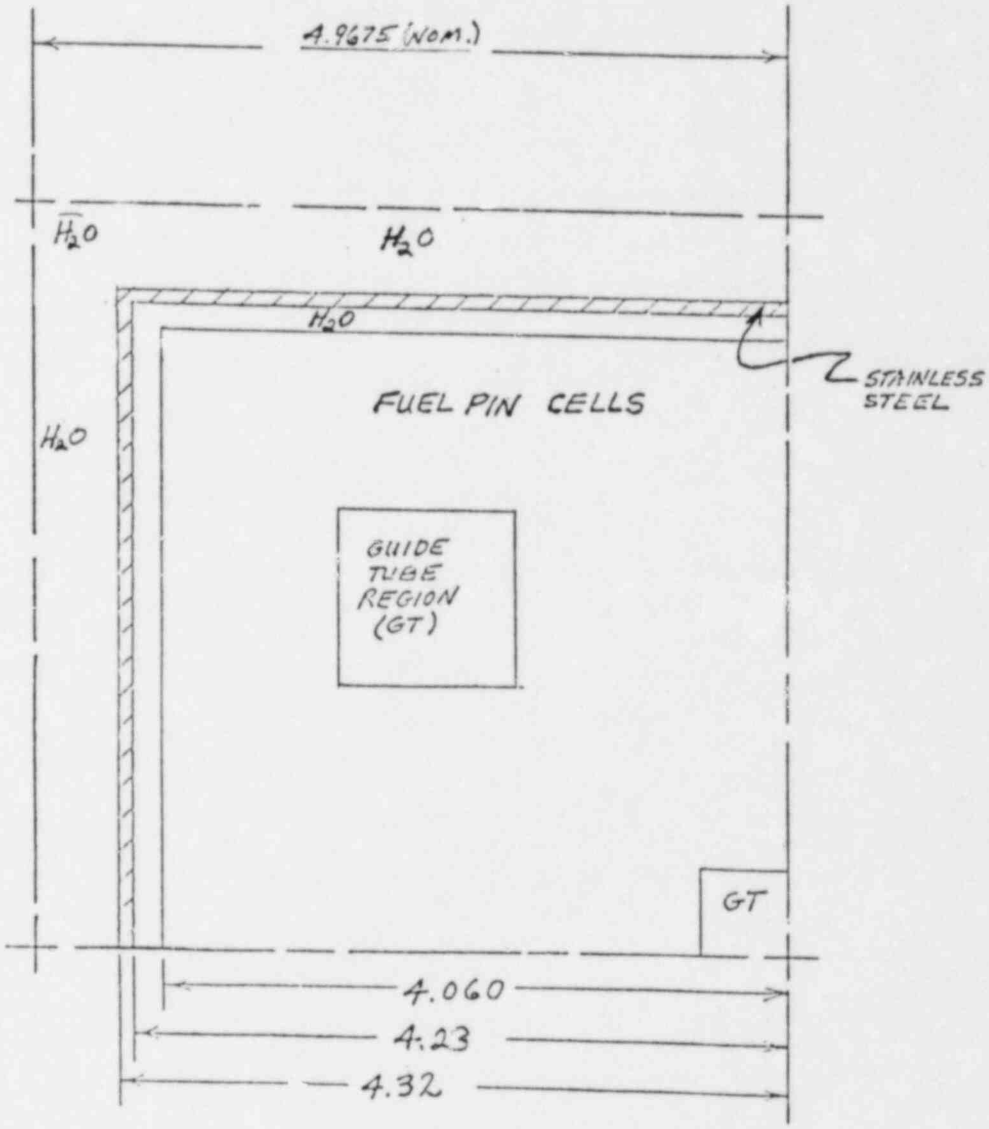


FIGURE 4.3-1
FORT CALHOUN SPENT FUEL RACK BASIC CELL
FOR REGION 2 CRITICALITY CALCULATIONS



Note: All dimensions in inches.

FIGURE 4.3-2

FORT CALHOUN FUEL ASSEMBLY k∞ AS A

FUNCTION OF FUEL ASSEMBLY EXPOSURE

(Minimum Fission Product Poisoning, No Boron, T = 68°F)

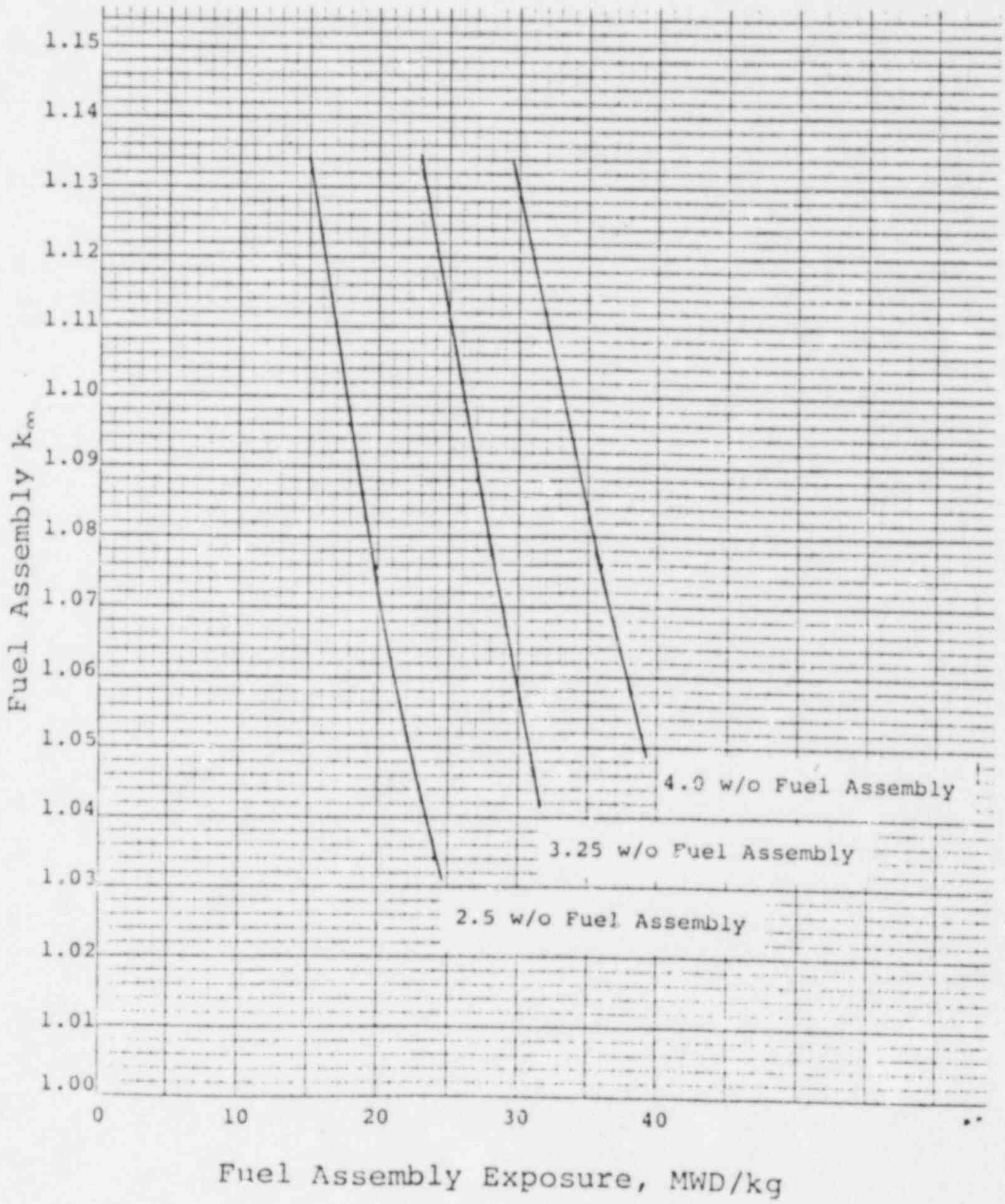


FIGURE 4.3-3

FORT CALHOUN REGION 2 K_∞ AS A FUNCTION OF FUEL ASSEMBLY k_∞ FOR SEVERAL INITIAL ENRICHMENTS

(No Boron, T = 20°C)

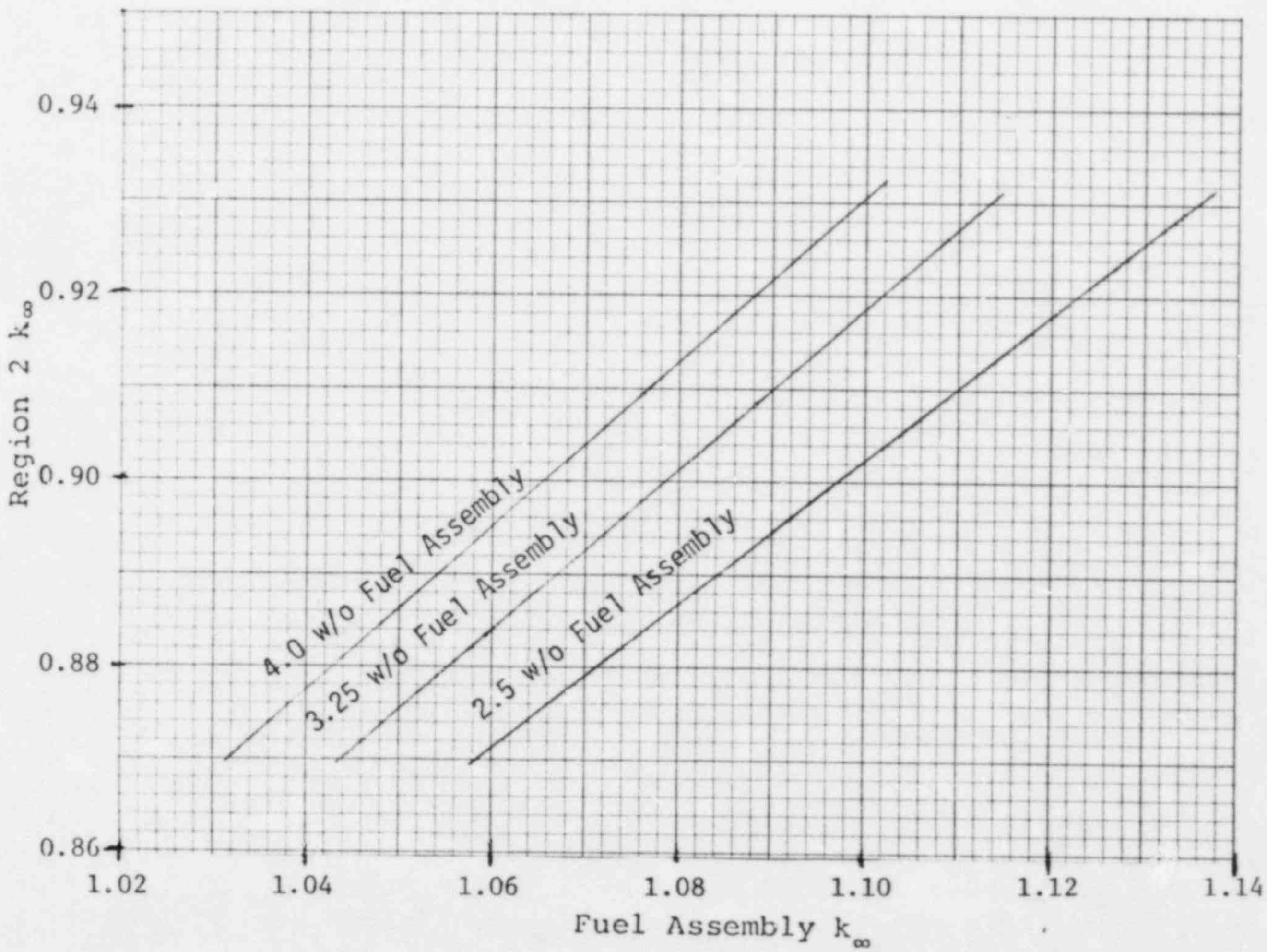


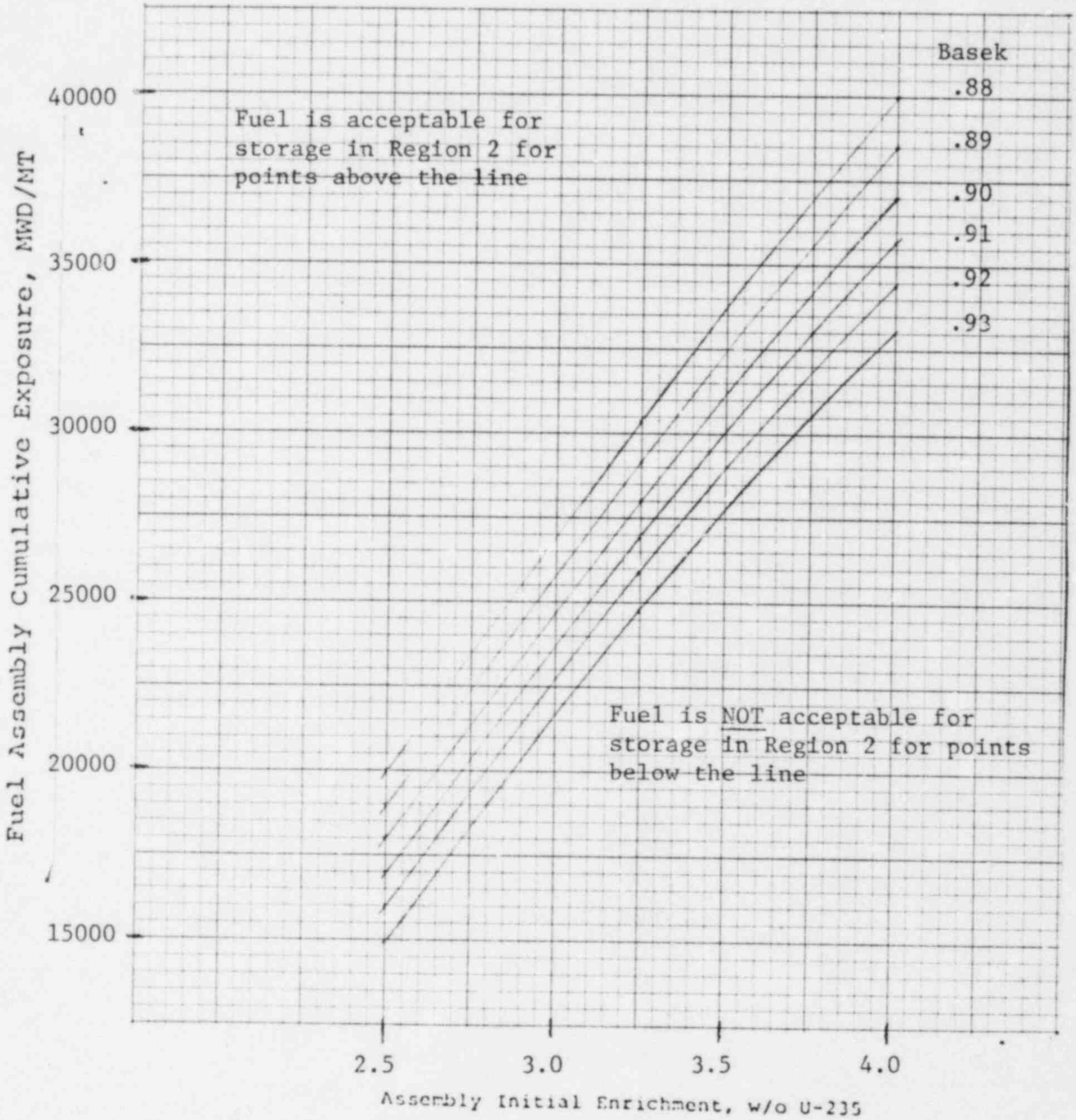
FIGURE 4.3-4

STORAGE CRITERIA FOR FORT CALHOUN REGION 2

Minimum Required Fuel Assemly Exposure

As a Function of Initial Enrichment

To Permit Storage in Region 2



*This is the Base Region 2 k_∞ to which all tolerances and calculated uncertainties must be added before final comparison against licensing criteria. From Table 9 it is observed that this total Δk is < 0.04 Therefore, as an example, if the licensing criteria dictated a maximum Region 2 k_∞ of 0.95 then the Base k_∞ must be ~ 0.91 for any combination of initial enrichment and exposures as identified from this figure.

FIGURE 4.4-1

FORT CALHOUN SPENT FUEL RACKS
CRITICALITY SAFETY ANALYSIS
EFFECT OF FUEL ENRICHMENT

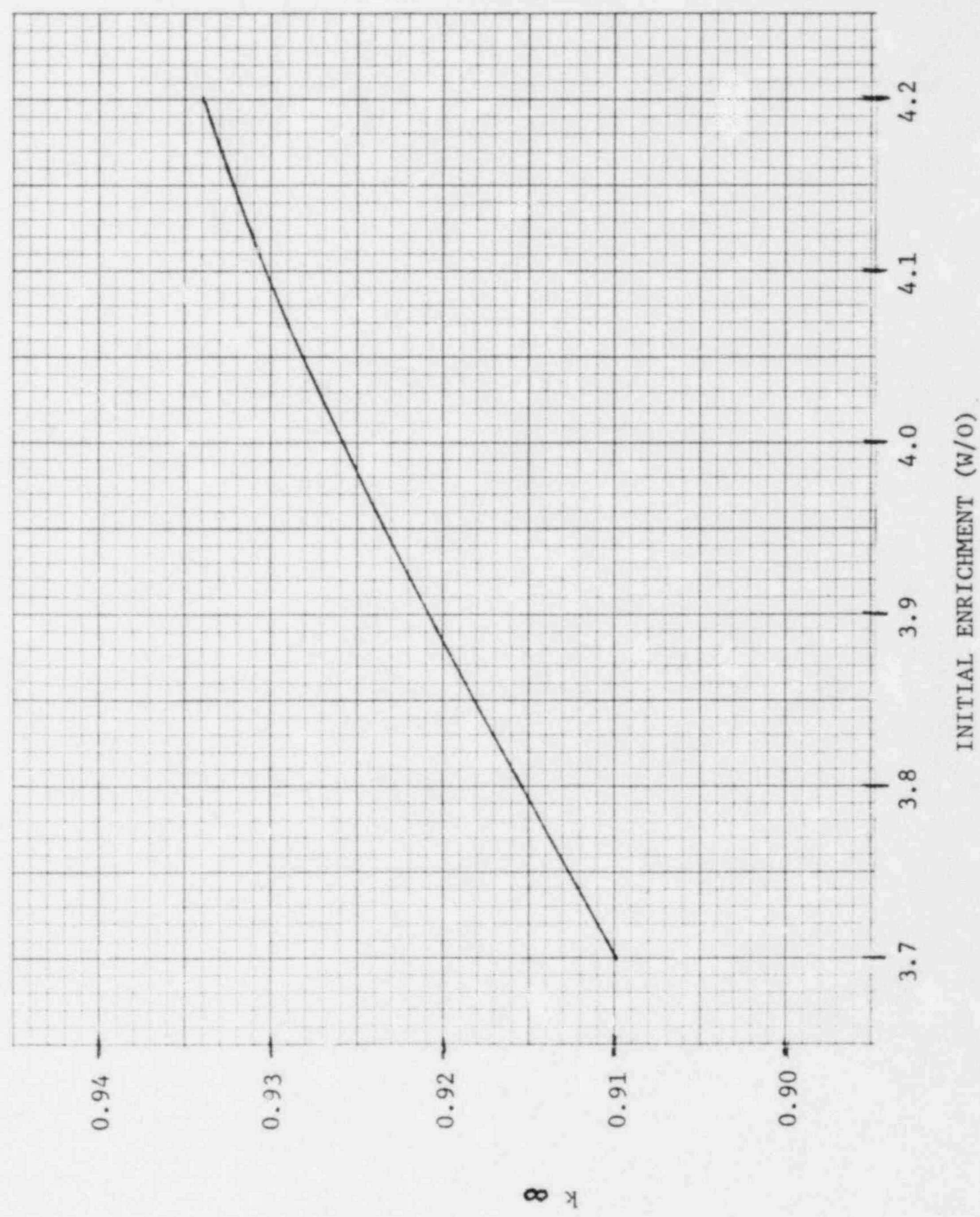


FIGURE 4.4-2

FORT CALHOUN SPENT FUEL RACKS
CRITICALITY SAFETY ANALYSIS
EFFECT OF B10 LOADING IN BORAFLEX

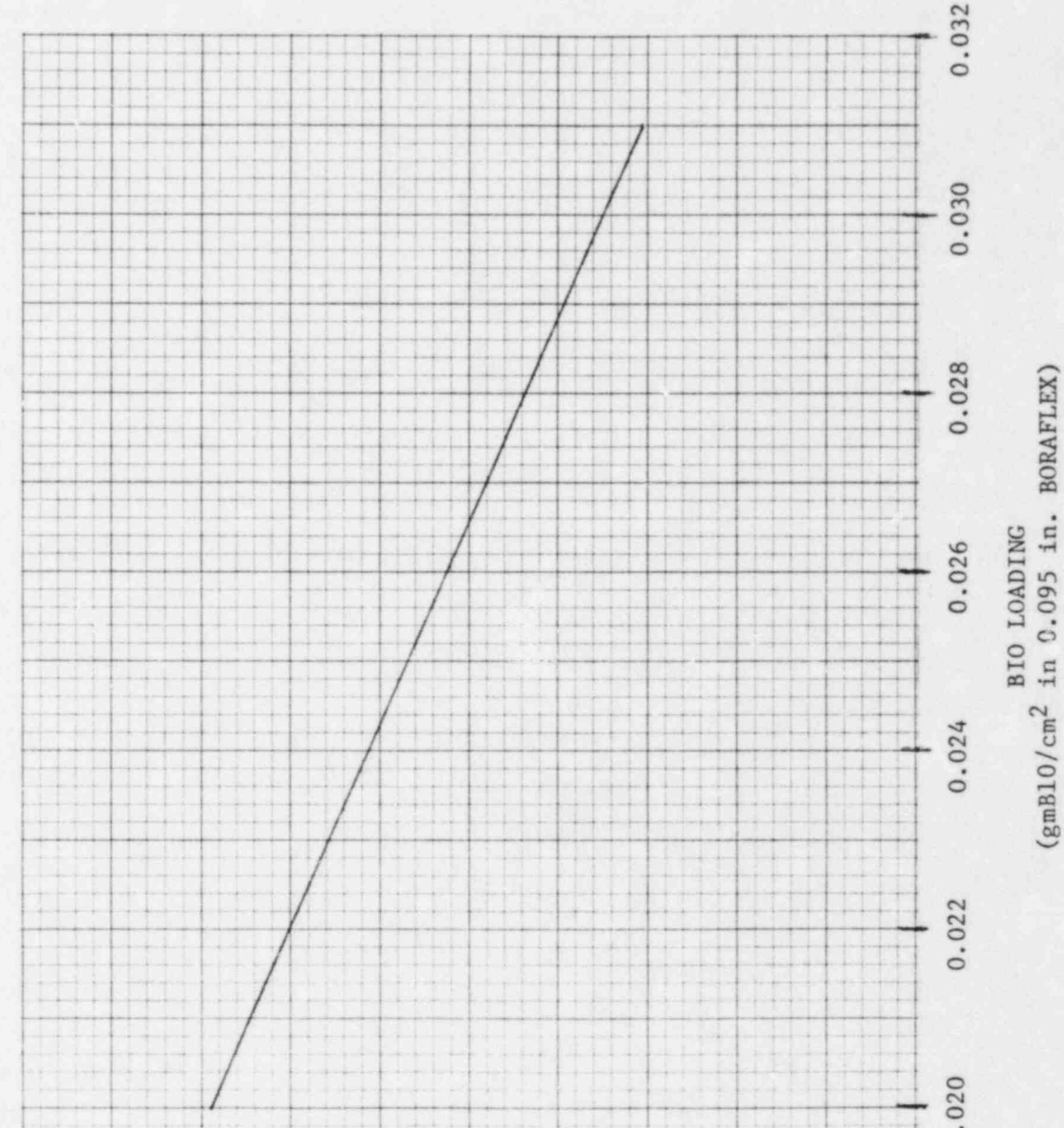


FIGURE 4.4-3

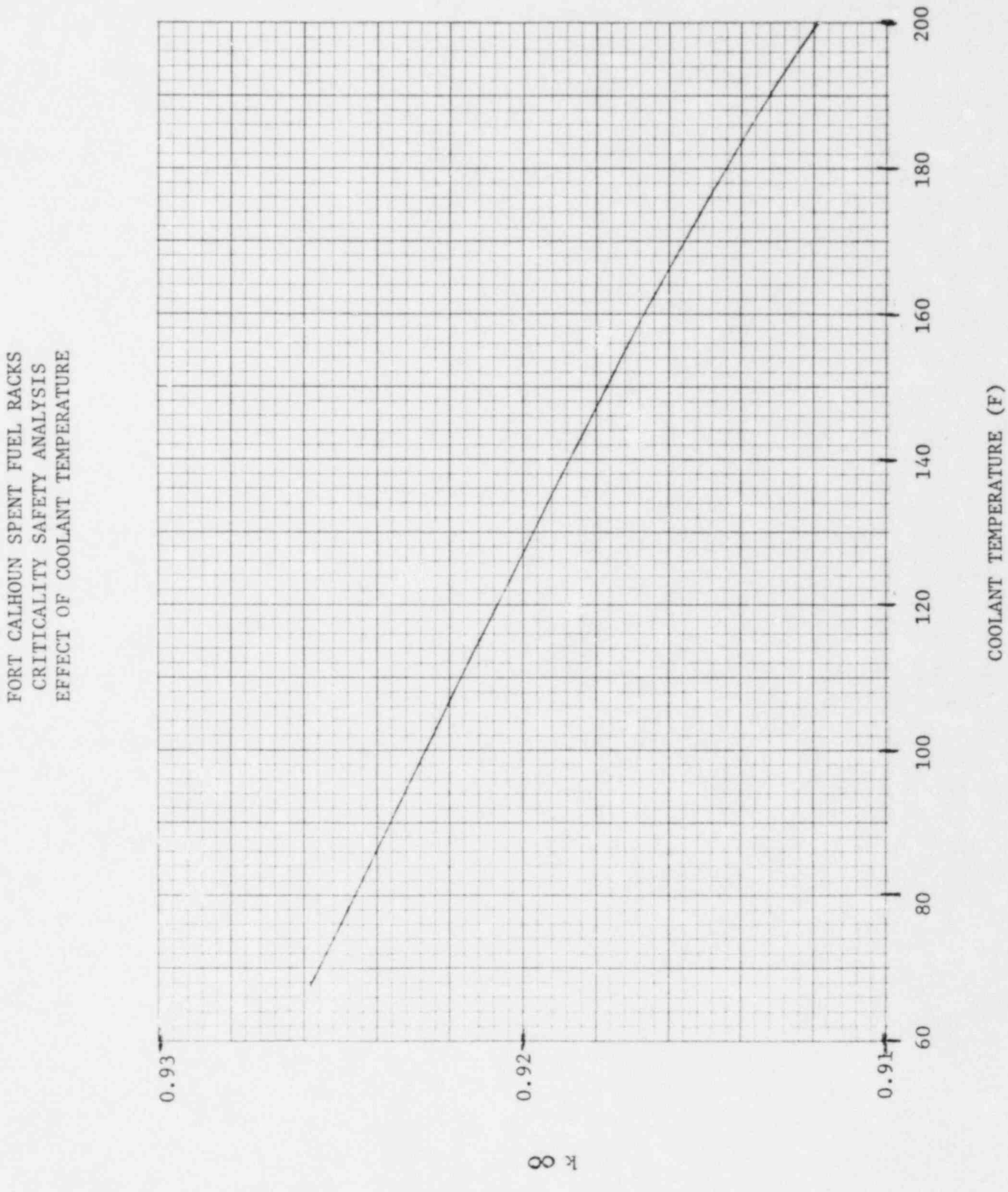

FORT CALHOUN SPENT FUEL RACKS
CRITICALITY SAFETY ANALYSIS
EFFECT OF COOLANT TEMPERATURE
COOLANT TEMPERATURE (F)
60
80
100
120
140
160
180
200
0.93
0.92
0.91
k_∞

FIGURE 4.4-4

FORT CALHOUN SPENT FUEL RACKS
CRITICALITY SAFETY ANALYSIS
EFFECT OF COOLANT DENSITY

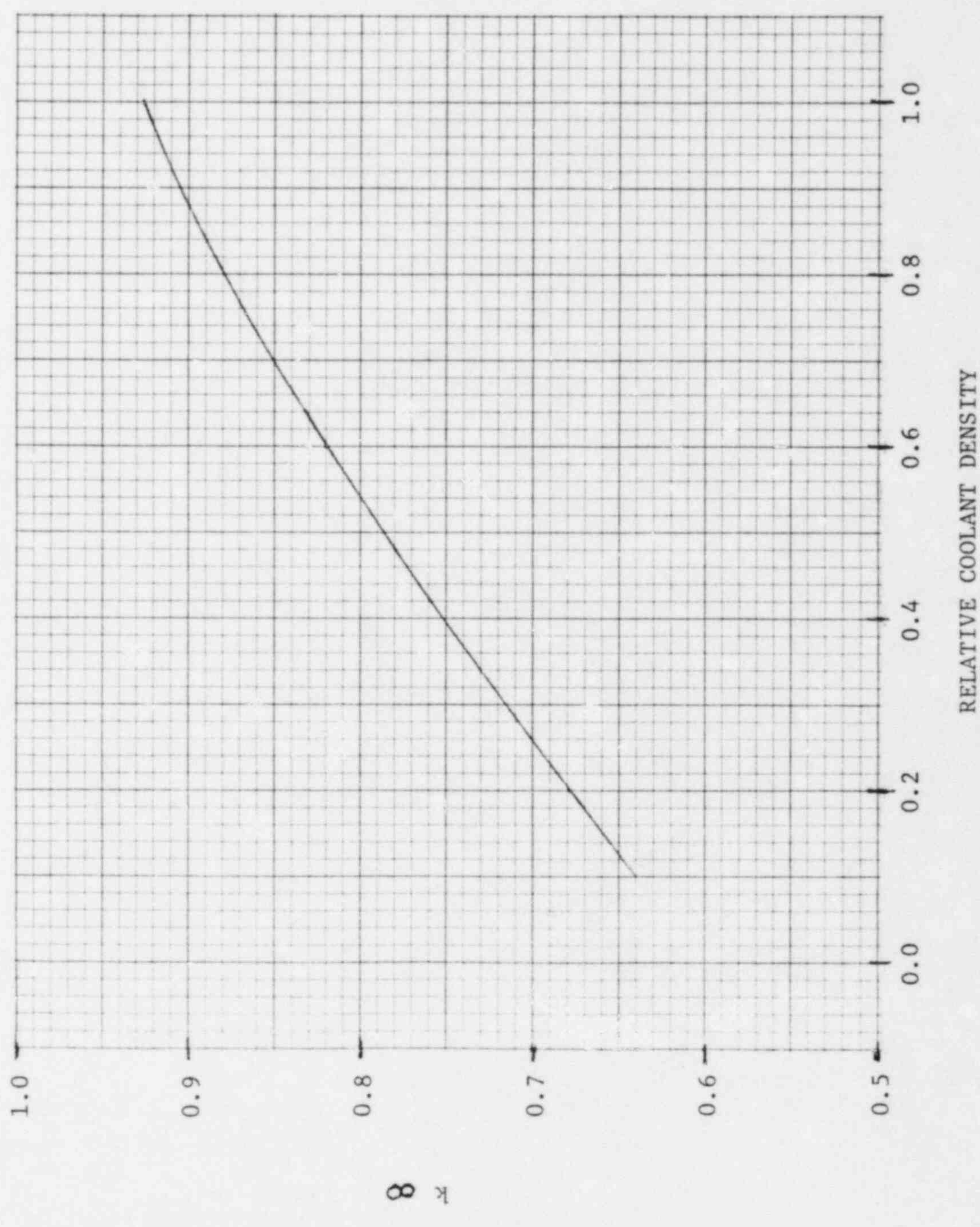


FIGURE 4.4-5

FORT CALHOUN SPENT FUEL RACKS - REGION 2
CRITICALITY SAFETY ANALYSIS
EFFECT OF COOLANT TEMPERATURE

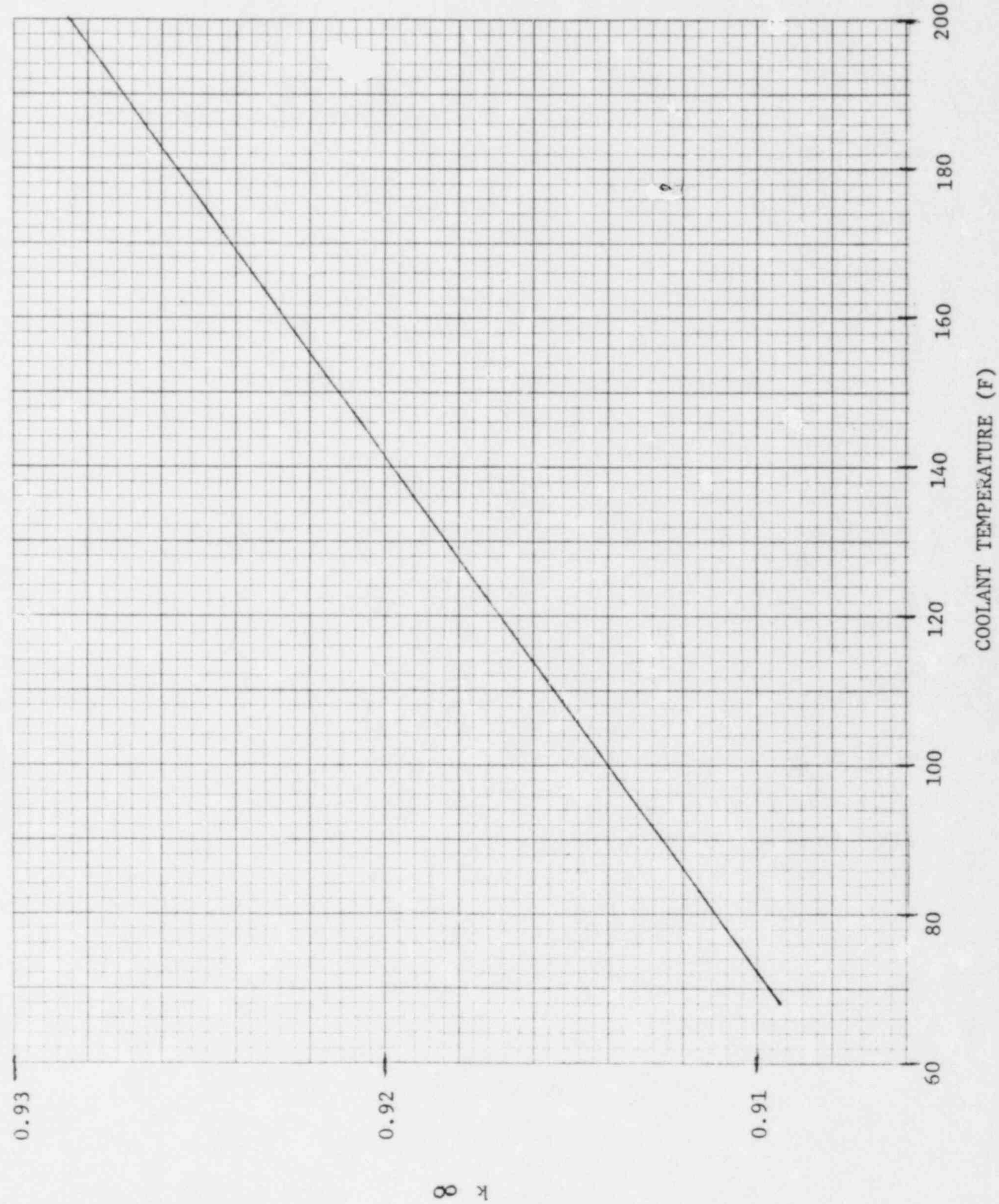
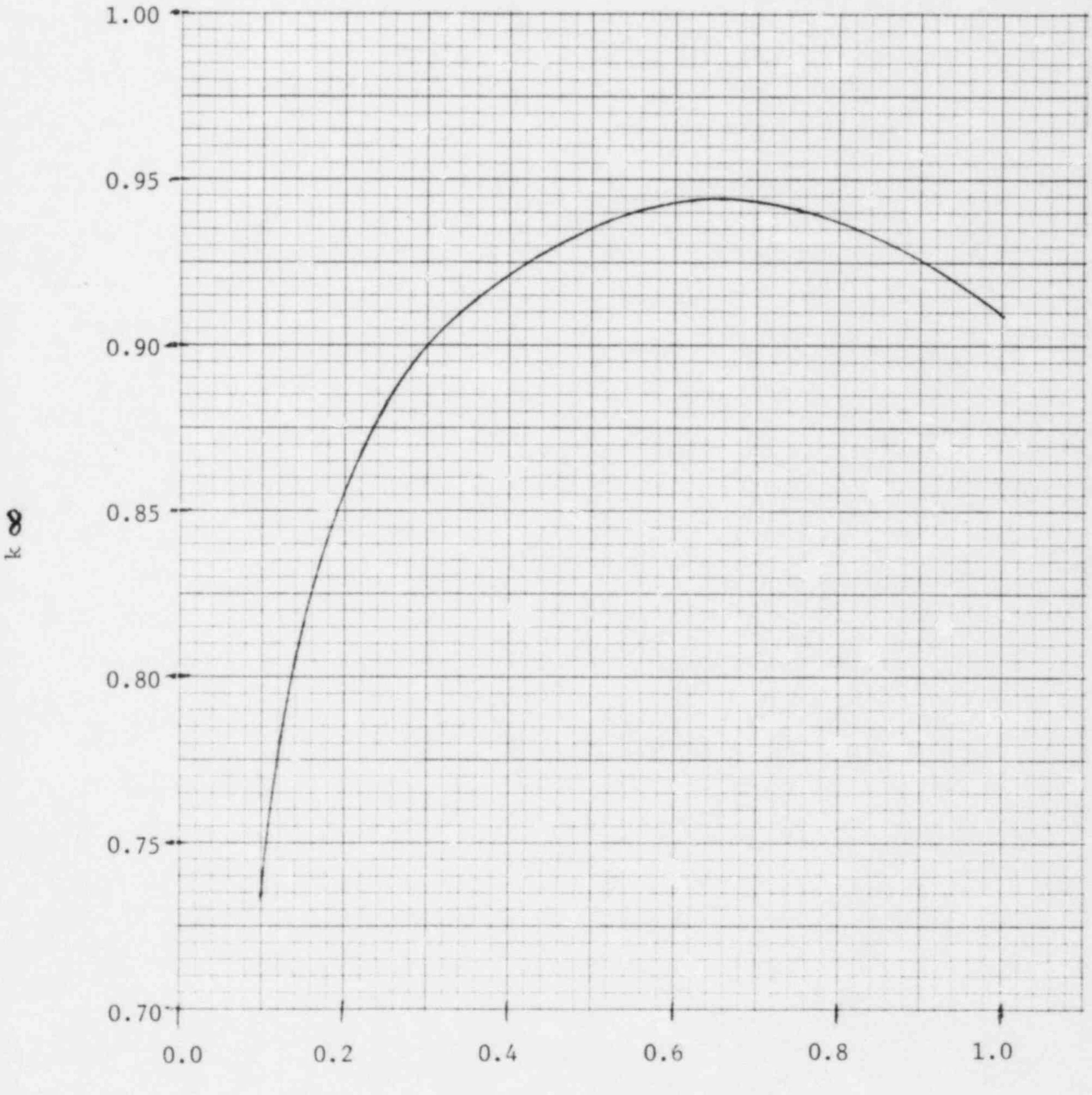


FIGURE 4.4-6

FORT CALHOUN SPENT FUEL RACKS - REGION 2
CRITICALITY SAFETY ANALYSIS
EFFECT OF COOLANT DENSITY



5.0 THERMAL-HYDRAULIC EVALUATION

This section describes the results of the analysis of the performance of the spent fuel pool cooling system. Although it would not be necessary to implement disassembly and compact storage of fuel until about 1994 and currently there is no firm commitment to such implementation, the adequacy of the spent fuel pool cooling system was evaluated assuming conservative decay heat loads corresponding to the ultimate storage capacity of the pool.

A thermal hydraulic analysis had been performed by Wachter Associates for the Wachter design for the previous submittal. This analysis is based on the GCA/PaR design. A major difference in this analysis is that Wachter Associates calculated decay heats using the ANS 5.1 criteria[2,3], this analysis uses the NRC Branch Technical Position ASB 9-2 to calculate the decay heat. Using the NRC criteria this analysis yields higher decay heat, and is more conservative.

5.1 DECAY HEAT LOADS FOR THE SPENT FUEL POOL

Table 2.0-1 of this document presents the expected discharge schedule for Fort Calhoun through year 2012 and cycle 27. Since the operating license for Fort Calhoun expires in 2008, this schedule should provide an upper limit estimate of fuel storage requirements. The cumulative number of assemblies is 1168 for discharges of 44 fuel assemblies every 1 1/2 years for two consecutive cycles starting with cycle 7, and 45 fuel assembly discharges are assumed every third cycle for cycles 9, 12, ... 27. Note cycle 7 listed in Table 2.0-1 indicates 28 fuel assemblies actually being discharged. However, for a conservative decay heat load calculation a worst case was used in which 44 fuel assemblies were assumed to be discharged. The normal refueling heat load will be based on this projection.

Maintaining full core discharge (FCD) capability, the normal storage limit based on 729 locations and assuming DCS would be approximately 45 + 2(596 - 45) = 1147; therefore, the 1168 storage limit (1301 with FCD) is a reasonable and conservatively high limit to impose here.

Although the current authorized reactor power level for Fort Calhoun is 1500 MW_{th}, decay heat loads are compiled based on a reactor power level of 1550 MW_{th} or $P_0 = 39.8 \times 10^6$ BTU/hr (11.65 MW) per fuel assembly. An exposure time of 1250 effective full power days (30,000 hrs) was selected based upon an assumed discharge burnup of 40,000 MWD/MTU.

Decay rates are found using the latest revision of the NRC BTP ASB 9-2. An uncertainty factor of 10% was used in the calculations. This analysis also covers the heat generated from heavy element decay, U-239 and Np-239. It should be noted that U-239 does not contribute at decay rates greater than 72 hrs, Np-239 does contribute to the total decay heat, as noted in Table 5.1-3.

A comparison of the decay power fraction (P/Po) versus decay time (ts) for 24,000 hrs., and 30,000 hrs., irradiation times (To) was provided in the previous licensing submittal. A comparison of the decay powers for the different irradiation times indicated that a To of 30,000 hrs., gave a higher decay heat load, thus an irradiation time of 30,000 hrs. was chosen when calculating the decay heat using the NRC criteria for this submittal.

The decay heat power fraction (P/Po) versus decay time is given in Figure 5.1-1 which includes the heavy element decay heats.

For normal refueling conditions, the minimum cooling time for spent fuel discharged to the fuel pool is 3 days or 72 hours, based on Section 9.6 of the Fort Calhoun USAR. Table 5.1-1 summarizes the assumed cycle histories which yield the highest expected normal refueling residual heat load. The most recent group of 45 fuel assemblies which was cooled 72 hours generated slightly less than 7.7×10^6 BTU/hr. With the spent fuel pool filled to capacity, the decay heat output would be approximately 11×10^6 BTU/hr. As noted by the arrow (→), the differential increase between compact and non-compact fuel storage is only 1.12×10^6 BTU/hr or about 10% of the design total. As the minimum t_s is changed from 72 hours (3 days) to 336 hours (14 days), the total cumulative heat load drops from 11×10^6 BTU/hr to 7.4×10^6 BTU/hr.

Assuming fuel can be moved as early as 72 hours (3 days) after shutdown and that it takes approximately 1/2 hour to move one fuel assembly from the reactor to the fuel pool, the minimum cooling time for FCD is about 6 days or 144 hours. To maximize the total heat load for full core discharge conditions, a history was chosen that maximizes the decay heat loads from the freshest 1/3 core (Cycle 30 with the shortest irradiation time in the reactor) plus the most recent 1/3 core from a normal refueling discharge already stored in the pool. Evaluations of this To, t_s variation indicate that the maximum heat load is very broad, and 90 days or 2200 hrs was then chosen as the minimum To. Thus, for the remaining 2/3 core, a To of 12,000 and 22,000 (Cycles 29 and 28, respectively) effective full power hour exposures were assumed (i.e., 10,000 hrs per cycle).

Reactor down-time was ignored for the 1/3 core already stored in the pool (i.e., t_s =2200 hrs for cycle 27). Table 5.1-2 presents a summary of the full core discharge decay heat load at the expected design conditions. Table 5.1-1 results were used for cycles 1 through 25. At the expected cooling system design limit (144 hrs or 6 days), the heat load is 22 x 10^6 BTU/hr. With two weeks as a minimum decay time, the total heat load drops off to 14.56 x 10^6 BTU/hr.

Based on Tables 5.1-1 and 5.1-2 the design heat load limits imposed on the Spent Fuel Pool Cooling System are 11 x 10^6 BTU/hr (normal refueling conditions) and 22 x 10^6 BTU/hr (full core discharge conditions).

5.2 BULK FUEL POOL AND HEAT EXCHANGER TEMPERATURES

The spent fuel pool cooling system is described in Section 9.6 of the Fort Calhoun USAR. The flow path in this system is such that heated water exits the fuel pool at the strainer located at elevation 1034' at the northeast corner. Another alternate suction path exists down in the pool at el. 996'-6". Two storage pool circulation pumps, in parallel, (combined flow of 1800 gpm) pump borated spent fuel pool water (SFPW) through the tube side of the storage pool heat exchanger and return the cooled water to the fuel pool. This 8" main line terminates with the cooling sparger, which is mounted on the pool floor (elevation 995.5') and traverses the width of the pool at the south end. Forced circulation is therefore northward and upward through the storage racks and spent fuel. This sparger is provided with a 1/2" diameter hole drilled into the pipe to prevent siphoning. Component cooling water (CCW) at a flow rate of 1160 gpm and a maximum inlet temperature of 90°F enters the shell side of the storage pool heat exchanger. Transferred heat is carried away to the raw water system and then to the Missouri River. The SFP system is single active failure proof, and as such proper cooling of the SFP is provided during normal or refueling conditions.

The design specifications of the heat exchanger (AC-8) are summarized in the USAR, Section 9.6. For purposes of determining the fuel pool bulk temperature above the racks, the heat loads summarized in the previous section can be used in conjunction with the heat exchanger effectiveness based on design point characteristics. Since increased temperatures will improve heat transfer rather than hinder it, this method will overestimate temperatures for heat loads exceeding that defined by the design point performance ($q_{dp} = 9 \times 10^6$ BTU/hr).

For the design point, the following data apply:

Fluid	"c" Shell Side (CCW)	"h" Tube Side (SFPW) Flow
Rate (gpm)	1160.	1800.
Inlet Temp (F°)	90.	110.
Outlet Temp (F°)	105.5	100.

Ignoring heat losses in the piping and assuming natural circulation will keep the pool coolant mixed, the fuel pool bulk temperature above the racks can be taken as equal to the tube side inlet temperature, T_h in. For the design conditions of interest, the following temperatures have been calculated:

Conditions	Heat Load (10^6BTU/hr)	T_h in (°F)	T_h out(°F)
Normal Refueling	11	114.5	102.3
Full Core Discharge	22	138.9	114.5

These results conservatively ignore slight improvements in the heat exchanger effectiveness (and overall heat transfer coefficient) for heat loads which exceed the design point performance value of $q_{dp} = 9 \times 10^6$ BTU/hr.

Under these same conditions, the variation of pool temperatures with minimum cooling time was determined. As shown in Figure 5.2-1, the normal refueling pool temperature limit is less than 120°F for any realistic cooling time ($t_s \geq 3$ days). For full core discharge, the pool temperature drops below 120°F when cooled about two weeks ($t_s > 16$ days). For a higher limit of 150°F, a minimum cooling time before discharge of only about four days is required.

It is also possible to make use of the shutdown cooling system to provide alternate or supplemental cooling of the spent fuel pool during a full core discharge. This system would use low pressure safety injection pumps and one shutdown heat exchanger. With approximately triple the heat removal capacity of the normal spent fuel cooling system, this system will be available to insure that the pool temperature limits stay well below 150°F for full core discharge. This system can only be used when the reactor is in a cold shutdown condition.

5.3 FUEL POOL HEAT-UP FOLLOWING LOSS OF COOLING

For fuel pool heat-up transients, it has been assumed that the pool temperature limits are below 120°F and 150°F for normal refueling and full core discharge conditions, respectively. Section 5.2 demonstrates that the SFP cooling system can be used to maintain the SFP temperature below 120°F. For a Full Core Discharge, shutdown cooling can be used or can supplement the SFP cooling system to keep the pool temperature below 150°F.

Initiating cooling using the shutdown cooling system will be possible within 2 hours after alarms notify the operators that the pool temperature limits are exceeded. The SFP alarms at 150°F. Accordingly, a two-hour pool heatup transient is analyzed in this section. The case where cooling can be restored (for either cooling mode) within two hours is also bounded by this analysis.

It was conservatively assumed that the pool water heats up adiabatically without accounting for evaporation losses at the pool surface or heat losses through the concrete walls and piping exterior to the pool. The previously established limits of 11 x 10^6 BTU/hr, 120°F for normal refueling and 22 x 10^6 BTU/hr, 150°F for the full core discharge were used. Reductions in heat loads and temperatures during the transient were conservatively ignored.

The thermal capacity, ρC_p, was estimated to be 61 BTU/ft^3 - °F for the pool water, and the pool volume is approximately 215,000 gallons. The total thermal capacity is thus, 1.75 x 10^6 BTU/°F.

Starting at 120°F, the pool will reach 132°F in 2 hours for normal refueling conditions. For full core discharge conditions starting at 150°F, the pool temperatures will reach 175°F in 2 hours.

If the pool is assumed to reach boiling before cooling can be restored, it will take: 1) 14.7 hrs (normal refueling) and 2) 4.9 hrs (full core discharge) for boiling to occur. Water would boil away at approximate rates of 11,340 lbm/hr and 22,680 lbm/hr for normal refueling and full core discharge conditions, respectively.

Depending on the degree of subcooling, make-up water can be added to compensate for these low rates of boil off to maintain the pool water level and keep the pool surface at or below boiling. Normally, make-up water is provided by the fuel transfer canal drain pumps from the safety injection and refueling water storage tank. Fire hoses could provide make-up water at these modest rates should the normal source malfunction. The radiological consequences of a boiling pool are discussed in Section 8.5, with regard to compliance to 10 CFR 100.

5.4 NATURAL CIRCULATION COOLING OF THE HOTTEST FUEL ASSEMBLIES

A natural circulation cooling analysis was performed for the Wachter design, and it was concluded from that analysis that adequate cooling would be provided using the Wachter design. This analysis used the same methods and assumptions in the previous analysis, performed by Wachter, except the analysis is applied to the PaR design. This is justifiable since there are only minor variations in the dimensions between the two designs. The one variation is in the bottom inlet grid of each fuel cell. The PaR design provides a grid instead of a plate with a hole. This provides approximately six (6) times the inlet flow area compared to the Wachter design. Also, the PaR design varies from the Wachter design in that the racks are only 4.25" off the pool floor instead of 5.00" as in the previous design. The affect of these variations is explained later in this section.

To demonstrate that the fuel is locally cooled and the fuel pool coolant mixes to a uniform temperature above the racks, a conservative natural circulation cooling analysis of the hottest fuel assemblies was performed. The hottest spent fuel assemblies would be those most recently discharged from the reactor (45 fuel assemblies for normal refueling, 133 for full core discharge) and were conservatively assumed to be located at the extreme end of the pool. But in actuality the hottest fuel assemblies will be discharged into the poisoned region (Region 1), which is located next to the cooling spargers. The fuel is discharged here prior to moving to Region 2. Thus the actual pool temperatures would be lower than those calculated using the extreme case of locating the fuel furthest from the sparger. Conservatively, the module racks at the north end of the pool are selected since they can accommodate 117 fuel assemblies. However, to be conservative 133 fuel assemblies were assumed to be stored in these racks, thus giving a higher decay heat.

The flow path chosen is the longest conceivable under rack path ($\sim$30 feet) originating at the south end, flowing north under the racks and driven (by density variations) up through the hottest fuel channels. Forced flow from the sparger, possible down-flow through colder fuel and channels, and down-flow along the pool walls, except the south wall was ignored.

For normal refueling conditions, each of the 45 fuel assemblies was calculated to have a thermal power of .244 x 10^6 BTU/hr. The total decay heat for 45 fuel assemblies was calculated to be 11 x 10^6 BTU/hr (from Section 5.1). For full core discharge (133 fuel assemblies) at 48.4 kw = 7 .165 x 10^6 BTU/hr was calculated to be the individual decay heat per assembly (144 hr decay time, Section 5.1). The total decay heat was therefore calculated to be 22 x 10^6 BTU/hr. Although some gamma heating will occur outside the fuel, all decay energy is conservatively assumed to be absorbed in the fuel for this evaluation.

The coolant temperature rise in the hottest assemblies was calculated from a simple energy balance and by equating the pressure losses of the flow path to the natural circulation driving pressure.

The driving head was conservatively estimated by using the height of the spent fuel boxes. Pressure losses were conservatively calculated for the bottom grid of the storage box, the lower and upper fuel assembly end fittings, the fuel assembly spacer grids, and the bare fuel rod friction. Pressure losses affecting the under rack flow were all conservatively calculated and added to those of the fuel assembly.

The resulting calculations for the coolant temperature rise indicated a weak dependence on the combined pressure loss coefficient, with a stronger dependence on the decay heat load and flow areas. This was documented in our earlier submittal with the Wachter design, it is also true for the PaR design, especially since the PaR design provides a larger flow area for the inlet to the fuel channels. The inlet flow area is about 6 times as great as the Wachter design, however the under rack clearance is only 4.25" compared to 5" in the Wachter design. The under rack clearance was the factor which determined how much the temperature change would vary. The temperatures calculated in the previous submittal based on the Wachter design were found to be lower than those calculated in this submittal due to the fact that the under rack flow areas were different.

Fluid properties were used at 120°F for normal refueling and 150°F for FCD, the coolant increases were found to be

ΔT_h = 39.1°F Normal Refueling

ΔT_h = 50.3°F Full Core Discharge

The coolant increases varied by 5°F and 8°F from the Wachter analysis for the above cases, respectively.

ΔT_h is the coolant change in temperature in the hot channel.

The ΔT's calculated above were based only on a natural circulation analysis and did not take into account the forced flow from the spargers and flow down the four walls. The only mixing that occurs in this analysis is due to the imbalance in the fluid densities in the channels and above the racks. (The density difference between the hot and cold channels balanced the resistive losses due to friction, expansion contraction, and turns.)

The ΔT's calculated for a hot channel using a natural circulation technique are not much higher than the ΔT's calculated across the heat exchanger. It should be noted that the heat exchanger has a large forced flow heat transfer capacity, whereas, the pool was not analyzed for any forced flow. Since the ΔT's are not abnormally hign, cooling by natural circulation will provide adequate local cooling, and keep the pool well mixed to a nearly uniform temperature above the racks.

Peak clad temperatures were estimated by adding the calculated clad film drops (22°F for normal refueling and 16°F for FCD) to the maximum calculated coolant inlet temperature. With the heat exchangers in operation, the coolant inlet temperatures to the hottest assemblies is taken to be the average heat exchanger temperature as determined in Section 5.2.

The results, together with the maximum inlet and outlet coolant temperature are summarized in Table 5.4-1. Results are also given for the postulated 2 hour loss of cooling transient. Above the fuel racks, the local saturation temperature for all cases is below saturation. For the most extreme case (FCD with 2 hr. cooling loss), the clad could reach local saturation. Considering the severity of this case and the extreme conservatisms used in the model, the fuel racks will provide an acceptable cooling geometry for heat removal by natural convection.

5.5 GAMMA HEATING AND COOLING OF THE POISON CHANNEL AND INTER-CELL WATER

Some of the gamma decay energy will be absorbed in the poison channel and inter-cell water; therefore, considerations must be given to heat removal requirements for these locations.

For water to get into the inner channels, holes are cut into the cell walls at the bottom and provided on all four sides (3/4 in. Diameter). The coolant will enter these holes flow up through the channel and flow out of holes which are provided at the top of the channel on all four sides (3/4 in. Diameter). Calculations were done to insure that the orifice hole area is large enough to keep the inter-cell water temperature below the coolant temperature in the fuel box and the box wall and poison temperatures below the clad temperature. The flow area required in the bottom orifices under these constraints is typically .3 to .4 in^2 per channel (2 channels per fuel assembly). Using this as a basis, the flow areas for the flow racks can be checked. Each channel provides 1.77 in. (four - 3/4" holes per channel) in area for coolant flow. Thus the flow area provided is about 4 times that which is required.

The poison sheets although encased in stainless steel wrappers are not completely sealed. Water can come in contact with the poison and if any off gassing occurs it will not be trapped, and create bulges in the cell or the wrapper. Since the channel spaces are not sealed voiding will not occur, and since holes are provided at the top and bottom of each inner can to permit a definitive flow path for circulation of water in these spaces, it can be concluded that the inter cell water and poison sheets will be adequately cooled, for the proposed storage racks.

REFERENCES FOR SECTION 5.0

1. NRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling," Standard Review Plan, Section 9.2.5-8a, Rev. 1, 1978, (This was previously designated as APCSB 9-2 prior to a branch change. There are no technical differences between the two standards).

2. American National Standards Institute ANSI N210-1976, "Design Ojectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

3. "Evaluation of Spent Fuel Storage Racks for Fort Calhoun Including Disassembly and Compact Storage of Individual Fuel Rods", PLG and WAI, October, 1980.

4. Updated Safety Analysis Report, Fort Calhoun Unit 1.

TABLE 5.1-1

Spent Fuel Pool Projected Design Decay Heat Load Summary
Normal Refueling Conditions

Based on BTP ASB 9-2

Cycle Number(s)	Fuel Assemblies Discharged	Cumulative No. of FA's	Decay Time^{T}s	Cumulative Heat Load 10^6(BTU/Hr)
	45	45	72 hrs	7.68
26	44	89	13,000 hrs	8.06
25	44	133	3 yrs	8.28
24	45	178	4.5 yrs	8.45
23	44	222	6.0 yrs	8.61
22	44	266	7.5 yrs	8.76
21	45	311	9.0 yrs	8.91
20	44	355	10.5 yrs	9.04
19	44	399	12.0 yrs	9.18
18	45	444	13.5 yrs	9.31
17	44	488	15.0 yrs	9.44
16	44	532	16.5 yrs	9.56
15	45	577	18.0 yrs	9.67
14	44	→ 621	19.5 yrs	9.79
13	44	665	21.0 yrs	9.89

TABLE 5.1-1 (Continued)

Spent Fuel Pool Projected Design Decay Heat Load Summary
Normal Refueling Conditions

Based on BTP ASB 9-2

Cycle Number(s)	Fuel Assemblies Discharged	Cumulative No. of FA's	Decay Time T_S	Cumulative Heat Load 10^6(BTU/Hr)
12	45	710	22.5 yrs	10.00
11	44	754	24.0 yrs	10.10
10	44	798	25.5 yrs	10.20
9	45	843	27.0 yrs	10.29
8	44	887	28.5 yrs	10.38
7	44 (Note 4)	931	30.0 yrs	10.47
6	40	971	31.0 yrs	10.55
5	197	1168	32.5+ yrs	10.91

Notes:

1. Reactor operation at 1550 MW_{th}
   Power per fuel assembly = 11.65 MW = 39.8 x 10^6 BTU/hr = Po
2. Irradiation time To = 30,000 hrs, Burnup $\sim$ = 40,000 MWD/MTU.
3. Arrow (→) marks approximate limit for normal (non-compact) fuel storage.
4. 28 fuel assemblies were actually discharged at the end of Cycle 7. To be conservative in the analysis, 44 assemblies were assumed to be discharged.

TABLE 5.1-2

Spent Fuel Pool Projected Design Decay Heat Load Summary
Full Core Discharge Conditions
Based on BTP ASB 9-2

Cycle Number(s)	Fuel Assemblies Discharged	Cumulative No. of FA's	Decay Time τ_S	Cumulative Heat Load 10^6(BTU/Hr)
30	45	45	144 hrs	4.42
29	44	89	144 hrs	9.79
28	44	133	144 hrs	15.35
27	45	178	2200 hrs	16.95
26	44	222	15,000 hrs	17.29
25	44	266	3 yrs	17.50
14 - 24	488	754	4.5 - 20 yrs	
1 - 13	547	1301	20 yrs. plus	20.14

Notes:

1. Reactor operation at 1550 MW_{th}
   Power per fuel assembly = 11.65 MW = 39.8 x 10^6 BTU/hr = Po

2. Irradiation time To = 30,000 hrs, Burnup ≈ 40,000 MWD/MTU
   except for cycles: 30, t_o = 2200 hrs
   29, t_o = 12,000 hrs
   28, t_o = 22,000 hrs

3. Results from Table 5.1-1 used for Cycles 1 - 25.

TABLE 5.1-3

Heavy Element Decay Heat Normal Refueling

CYCLE	ASSEMB.	ts	P/PoU239	P/Po Np239	CUMUL. HEAT LOAD X10^6 BTU/hr
27	45	72 hrs	0	6.32×10^{-4}	1.131
26	44	13,000 hrs	0	0	1.131
25	44	3 yrs	0	0	1.131
1-24	1045	4.5 - 32 yrs	0	0	1.131

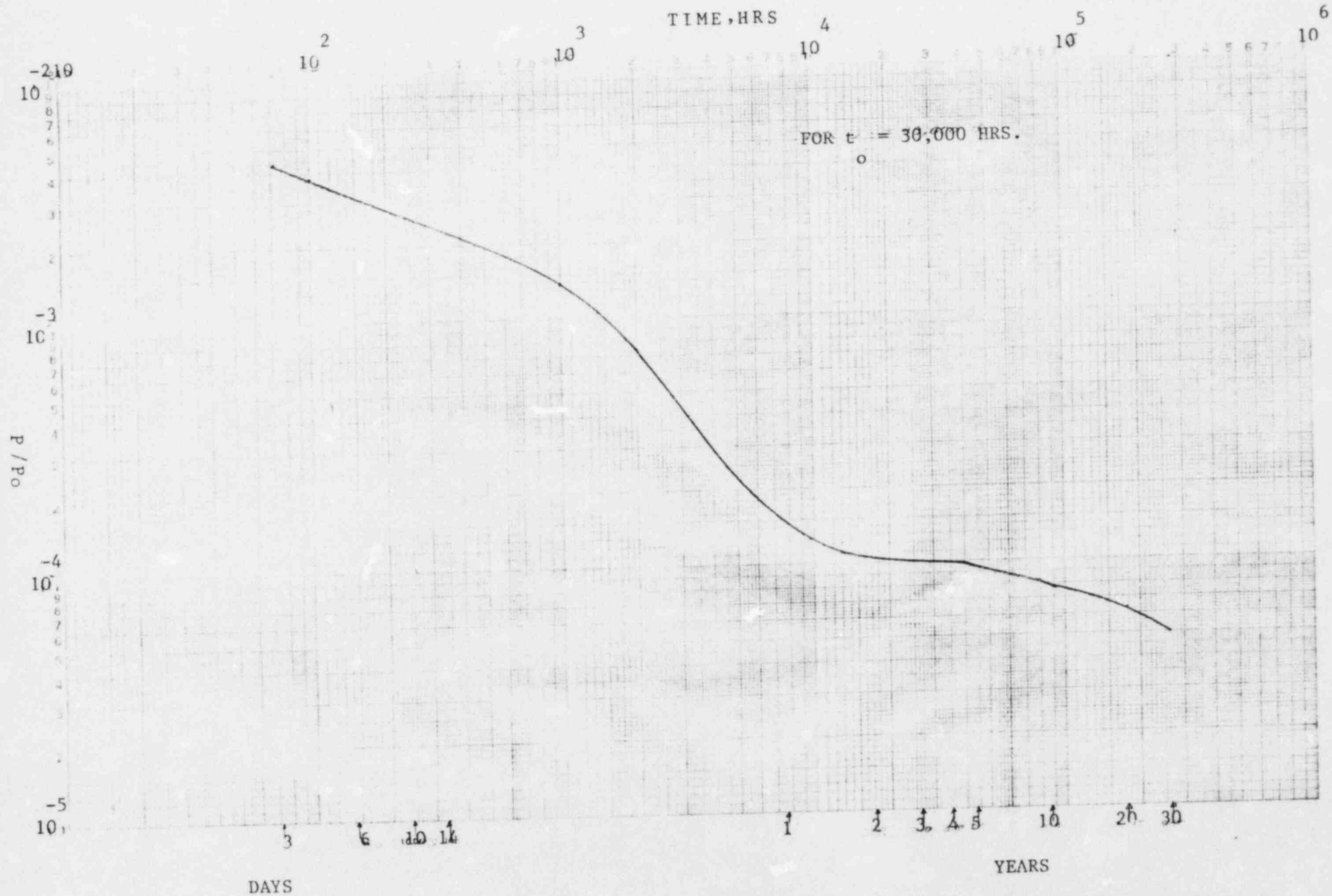
TABLE 5.4.1

Results of Natural Circulation Analysis

	Conditions	T_{in}(°F)	T_{out}(°F)	T_{clad}(°F)
Normal	With HX operation	108	147.2	170.5
Refueling	2 hr. loss of cooling	132.5	171.9	194.5
Full Core	With HX operation	126	176.5	192.1
Discharge	2 hr. loss of cooling	175	225.8	241*

Notes:

1. T_{in} = Coolant inlet temperature
2. T_{out} = Maximum coolant temperature in the hottest assemblies
3. T_{clad} = Maximum clad temperature
4. The fuel assembly thermal powers are taken to be 71.6 KW (.244 x 10^6 BTU/hr) for normal refuleing and 48.4 KW (.17 x 10^6 BTU/hr) for the full core discharge.
5. Above the racks, T_{sat} >240°F. The coolant will be subcooled for the extreme case (*). The clad could reach local saturation, but will not exceed the 244°F estimate.
6. 4.25" Clearance between the floor and the racks



FRACTION OF OPERATING POWER VS. COOLING TIME (t_c)

6.0 MECHANICAL, STRUCTURAL, AND MATERIAL EVALUATION

This section summarizes the criteria and analytical methods for mechanical and structural analysis which will be used to demonstrate that the spent fuel racks are designed and fabricated to meet and utilize the applicable portions of the Regulatory Guides, Safety Review Plans, and Published Standards listed in Table 6.1.

6.1 BASIS FOR LOADS AND STRESSES USED IN MECHANICAL AND STRUCTURAL EVALUATIONS

The following contains the loads, load combinations and design allowable stress to which the racks will be designed.

6.1.1 The loads considered in this design are:

- D - Dead Load of Racks
- L - Live Load due to the weight of fuel assemblies which shall be considered as varying from zero to full load and loadings corresponding to varying placement of the fuel assemblies in the rack shall be considered so that the most critical loads are obtained.
- E - Loads generated by operating basis earthquake (OBE) resulting from ground surface horizontal acceleration and vertical ground surface acceleration acting simultaneously.
- E' - Loads generated by safe shutdown earthquake (SSE) resulting from ground surface horizontal acceleration and vertical ground acceleration acting simultaneously.
- T - Thermal loads for water temperature at 150°F minimum water temperature is 40°F.

T' - Thermal loads for pool surface temperature at 212°F (240°F at rack elevation)

P - Lifting force of 4000 lbs applied to the top of the rack at any fuel bundle location.

H - Horizontal force of 1000 lbs applied at top of the rack at any fuel bundle location.

I - Impact load resulting from following conditions:

Condition 1 - Max. Impact load from Table 6.2 impacting on the middle of the top grid.

Condition 2 - Max. Impact load from Table 6.2 impacting on the corner of the top grid.

Condition 3 - Fuel Bundle drop from 15 inches above the rack free falling through an empty cavity and impacting the bottom grid.

Condition 4 - Inclined fuel bundle drop on the top of the rack.

Condition 5 - Gate drop from 52.2 in. above the rack impacting on the top of the rack

6.1.2 Load Combinations The following load combinations shall be satisfied:

1) D + L + H + T
2) D + L + P + T
3) D + L + E + T
4) D + L + E' + T'
5) D + L + T + I

Condition 1
Condition 2
Condition 3
Condition 4
Condition 5

6.1.3 The acceptance criteria for load combinations in 6.1.2 are:

Load Combinations	Limit
1) D + L + H + T	F_S
2) D + L + P + T	F_S
3) D + L + E + T	F_S
4) D + L + E' + T'	$1.6F_S$
5) D + L + T + I	$1.6F_S$
Condition 1	
Condition 2	
Condition 3	
Condition 4	
Condition 5	

NOTE: Local failure of the fuel support or the top grid impact interface is allowed. However, overall member stresses shall be limited to $1.60F_s$ and the configuration is such that k_{eff} will not exceed .95.

F_s = Allowable working stress
F_s = Calculated stress
F_y = Yield stress

6.1.4 Allowable stresses (For Stainless):

The allowable stresses are in accordance with ASME Boiler and Pressure Vessel Code Section III Appendix XVII, Subsection NF. This is interpreted as being identical to the AISC Steel Construction Manual.

The one-third increase in allowable stress for emergency conditions is not allowed. The increase in allowable stress is defined by the preceding paragraph 6.1.3.

6.2 WEIGHTS AND SPENT FUEL POOL FLOOR LOADING

All weights and loads used in the analysis are based on the conservative assumption that each storage location may eventually be used to store individual fuel rods according to the DCS proposal. The weights of such canisters are calculated based on the assumption that each such canister would contain twice the number of fuel rods in a single fuel rod storage area.

The largest rack is an 8 x 9 array of storage boxes, and this rack was used to develop weights and loads since it maximizes the pool floor loading value per single rack pedestal. This 72 unit rack has a dry empty weight of 15,800 lbs. and a submerged weight of 13,800 lbs. When fully loaded with DCS cannisters, these weights become 187,200 lbs. and 163,800 lbs., respectively.

The bearing plates which transfer the load of the racks to the fuel pool floor will be sized to meet the pool liner unit load limit of 500 psi. Including the accelerations resulting from the assumed maximum hypothetical earthquake or SSE, the maximum unit loading on the liner underneath the pedestal would be below 500 psi. The maximum overall floor loadings will be under the allowable stress, (D + L + E' + T')

The total weight on the fuel pool floor is 3,441,645 lbs., from racks fully loaded with DCS canisters (the equivalent of 1456 fuel assemblies), which is the number of fuel assemblies that could be stored if all were disassembled, plus the weight of the pool water. This corresponds to an average unit loading on the pool floor of 39.1 psi.

The overall pool floor loading associated with a canister drop (worst case drop accident) onto the storage racks (D + L + T + I) will be below the loading stress limit of a seismic event. The drop accident is discussed further below. In regard to thermal stresses the racks are freestanding with no fixed end thermal stresses resulting. The thermal concerns will be included by decreasing the material yield strengths for the applicable temperature level.

6.3 ANALYSIS OF FUEL DROP ACCIDENT

Analysis of the fuel drop accident that will be performed will guarantee that only minimal damage will occur to the rack, and there will be no significant effect on the subcriticality of the racks. A list of loads handled over the racks is found in Table 6.2. This table includes load weights, heights, and kinetic energy upon impact. All loads include the lifting fixtures.

Various drop accident analyses are being performed which address the effects of dropping the loads identified in Table 6.2 into the SFP. Postulated drop accidents will include but not be limited to 1) a straight drop on the top of the rack, 2) a straight drop through an individual cell all the way to the bottom of the rack, 3) an inclined drop on the top of a rack, and 4) one fuel canister impacting another fuel canister already in a storage cell. These postulated load drops will be reviewed under mechanical, material, and structural disciplines, to confirm compliance with the criteria listed in this document.

The most serious radiological consequences of dropping a spent fuel assembly into the spent fuel pool have previously been conservatively analyzed and shown to be acceptable,* also see Section 8.6. The consequences of this hypothetical accident are not changed by the installation or utilization of new spent fuel storage racks.

The OPPD Auxiliary Building crane which traverses over the spent fuel pool is prohibited from traveling over the spent fuel pool by electrical interlocks. Heavy loads can only be carried in the cask pit area, therefore, no heavy loads will be moved over the pool. The heaviest load that the crane can carry near the racks in the pool is the spent fuel pool gate. The results from a gate drop impacting the racks will be below the stress limits of load combination 5 condition 5.4 (Section 6.1.3).

Protection against a cask drop is assured by the Seismic Category 1, single failure-proof, Auxiliary Building crane, by the single failure-proof lifting device, and by the interlocks and administrative controls placed on the crane. The crane is provided with interlocks as described above. The cask is handled in the cask pit area, and is prohibited from traveling over the fuel racks. The re-rack program will not alter the cask handling procedures. The criticality and radiological consequences of a cask drop in the spent fuel pool are shown in Sections 4.6 and 8.5, respectively.

*"Omaha Public Power District's Fort Calhoun Station Stretch Power Environmental Assessment," Appendix A to Application for Amendment of Operating License for Fort Calhoun Station Unit No. 1, July 17, 1979.

6.4 MAXIMUM PILE LOADINGS AS A RESULT OF DCS OF SPENT FUEL AT FORT CALHOUN

The resulting pile loadings will be compared to the design loads in the following table:

Assumed Loads	Design (2) Basis Loads (KIPS)
Deadweight = D	360
D + OBE	580
D + SSE	610

The maximum pile loadings w. .ı full implementation of DCS will remain within the design basis pile loading. If DCS is not utilized and only fuel assemblies are stored in the racks, the maximum deadweight load is considerably reduced.

6.5 SPENT FUEL POOL STRUCTURE

The Spent Fuel Pool (SFP) floor is a 12'-0" thick concrete mat supported on steel pipe piles. The SFP wall thicknesses range from a minimum of 4'-0" to 5'-6" thick reinforced concrete. A stainless steel liner plate 3/32" thick provides a water tightness to the SFP walls and floor. There are no rooms below the spent fuel pool.

The (SFP) structure is classified as a Class I structure as defined in the Fort Calhoun USAR, Appendix F. The design criteria for Class I structure is discussed in Section 5.11 and Appendix F of the USAR. This design criteria states that the pool structure components were designed as per the requirements of the American Concrete Institute (ACI) Code 318-63. The allowable stresses and factor of safety for reinforced concrete as permitted by this Code for various load combinations were used in the des'gn of the Spent Fuel Pool Structure.

REFERENCES FOR SECTION 6.0

1. Fort Calhoun USAR, Section 5.7.

2. Gibbs and Hill Drawing 11405-S-1, Rev. 7, 5/6/68.

3. Gibbs and Hill Drawing 11405-S-71, Rev. 6, 4/16/69.

TABLE 6.1

Basis for Design Criteria for the Fort Calhoun Spent Fuel Storage Racks

1. United States Nuclear Regulatory Commission (USNRC)

a.	Reg. Guide 1.28	Quality Assurance
b.	Reg. Guide 1.13	Spent Fuel Storage Facility Design Basis, Rev. 2, Draft
c.	Reg. Guide 1.29	Seismic Design Class, Rev. 3, 1978
d.	Reg. Guide 1.92	Combination of Modes in Seismic Analysis, Rev. 1, Feb. 1976
e.	Reg. Guide 1.38	"QA Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water Cooled Nuclear Power Plants", Rev. 2, 1977.
f.	Reg. Guide 1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants, Rev. 1, Dec. 1973
g.	Reg. Guide 1.61	Damping Values for Seismic Design of Nuclear Power Plants, Oct. 1977
h.	Reg. Guide 1.31	Control of Ferrite Content in Stainless Steel Weld Metal, Rev. 3, April 1978
i.	Reg. Guide 1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports, Rev. 1, Jan. 1978
j.	SRP 3.7	Seismic Design, 1975
k.	SRP 3.8.4	Seismic Category 1 Structures, 1975
l.	SRP 9.1.2	Spent Fuel Storage, 1975

m. NRC Guidance on Spent Fuel Pool Modifications, Review and acceptance of Spent Fuel Storage and Handling applications (April 14, 1978 revised Jan. 18, 1979).

Table 6.1 (Continued)

Basis for Design Criteria for the Fort Calhoun Spent Fuel Storage Racks

2. Industry Codes and Standards

a.	ASME	Boiler and Pressure Vessel Code Section IX and Section III, Appendix I, VXII, and Article NF-4000, 1980 Edition (American Society of Mechanical Engineers).
b.	AISC	Steel Construction Manual AISC 8th Edition, December 1980 (American Institute of Steel Construction).
c.	ACI 318-71	Building Code Requirements for Reinforced Concrete. (American Concrete Institute).
d.	ASTM	ASTM Standards: A-240, A-666, A-564.
e.	ANSI N45.2	"Quality Assurance Program Requirements for Nuclear Facilities" 1977.
f.	ANSI N45.2.2	"Packaging and Shipping, Receiving Storage and Handling of items for Nuclear Power Plants", 1972, except Paragraphs 2.4 and 2.6.
g.	ANSI N210	Design Objective For Light Water Reactors Spent Fuel Storage Facilities at Nuclear Power Stations, 1976.
h.	ANSI N45.2.10	"Quality Assurance Terms and Definition," 1973.
i.	SNT-TC-1A	Recommended Practice for Personnel Qualification and Certification in Nondestructive Testing, American Society for Nondestructive Testing, 1975.

3. Federal Specifications (Standards)

a.	10 CFR 50	Code of Federal Regulations, Title 10, Part 50 Appendix A and B.
b.	10 CFR 20	Standards for Protection Against Radiation
c.	10 CFR 21	Reporting of Defects and Noncompliance

TABLE 6.2

KINETIC ENERGY FROM LOADS DROPPED OVER RACKS

Load	Weight	Height	Kinetic Energy Upon Impact on the Top of a Fuel Rack
Fuel Assembly	1210 lbs	15"	18,150 in-lbs
Fuel Assembly and Tool	1500 lbs	15"	22,500 in-lbs
Fuel Canister	2380 lbs	15"	35,700 in-lbs
Fuel Canister and Handling Tool	2680 lbs	15"	40,200 in-lbs
Control Element Assembly	70 lbs	196"	13,720 in-lbs
Handling Tools	300 lbs	196"	58,800 in-lbs

TABLE 6.2 (Continued)

KINETIC ENERGY FROM LOADS DROPPED OVER RACKS

Load	Weight	Height	Kinetic Energy Upon Impact on the Top of a Fuel Rack
Incore Detectors	100 lbs	189"	18,900 in-lbs
Control Rods	50 lbs	189"	9,450 in-lbs
Camera	25 lbs	189"	4,725 in-lbs
Gate	5776 lbs	52.2"	15,900 in-lbs/per fuel cell assumed 19 impacted.

7.0 SEISMIC EVALUATION

A seismic analysis of the spent fuel storage racks will be performed to demonstrate the ability of the supported equipment to remain fully functional during and after an assumed seismic disturbance. The adequacy of this equipment will be demonstrated by a complete mathematical analysis of the spent fuel racks, subjected to either the Operational Basis Earthquake (OBE) or the Safe Shutdown Earthquake (SSE). The analysis will be completed by March 15.

Based on the extensive seismic analysis summarized in the following sections, it can be concluded that the design limits derived from the seismic analysis, and used in the mechanical and structural evaluation, will be appropriately conservative for use in the design of the spent fuel storage racks based on a similar analysis performed for the J. M. Farley fuel racks.

7.1 STRUCTURAL ANALYSIS METHODS

The following is a brief description of the methods which will be used to structurally analyze the rack design. These methods have been licensed by PaR for several other rack projects. The free standing rack design is structurally qualified by a detailed time history (T.H.) and static analysis.

Simplified time history analyses are done at both 0.2 and 0.8 coefficients of friction (μ) conditions with 0, 1/4, 1/2, 3/4, and full eccentric fuel loading conditions. The low coefficient (0.2) is used to define maximum credible sliding displacement and the higher coefficient (0.8) is used to define the worst loading condition on a rack. These simplified analyses are done using PaR proprietary computer program RCKN1 and are further explained in Paragraph 7.2.

A detailed time history analysis using the ANSYS computer code is performed for the worst loading condition (as determined from RCKN 1) on a 3-dimensional double rack model (6x9 and 8x9). This detailed analysis will define the complete dead, live, and seismic stresses on the two racks modeled. The model and analysis method are futher explained in paragraph 7.3.

A static analysis is done on the detailed ANSYS finite element models for the following load cases:

a) Rack "dead" load ("D" loading, Para. 6.1.1)
b) fuel "live" load ("L" loading, Para. 6.1.1)
c) impact load ("I" loading, Para. 6.1.1)
d) fuel handling ("P" loading, Para. 6.1.1)

This model and method for analysis are further described in Paragraph 7.4.

The stresses resulting from the static analysis shall be combined per load combinations 1,2, and 5 of Paragraph 6.1.3. The stresses from the time history analysis are per load combinations 3 and 4 of Paragraph 6.1.3. The resultant member stresses are then compared against the member allowable stress limits (Ref. Paragraph 6.1.2 thru 6.1.3).

A summary table is then compiled showing the maximum rack stress interaction for each beam and plate element type.

NOTE: 1) The combining is done internally in ANSYS (See Figure 7.3-1).
2) The thermal loads will also be included in the Paragraph 6.1.3 load combinations by decreasing the materials yield strength at the applicable temperature level. No fixed end thermal stresses will be included since the proposed racks are free standing.

7.2 RCKN1 TIME HISTORY MODEL DESCRIPTION

This model (Figure 7.2-1) is used to determine the worst loading conditions based on the aforementioned eccentric fuel loading conditions and varying coefficients of friction. Fuel rattling and rack/rack interaction are not considered in this analysis. In this program the rack is idealized as a vertical beam connecting to a base via a torsional spring. This spring beam is sized to match the lowest horizontal rack frequency. In addition, this beam may be located eccentrically on the base to account for eccentric fuel loading. At each corner of this base, vertical gap springs are located. These springs take only compression loads allowing for rack uplift and rocking. At the lower left corner of the base a horizontal slider spring is located. This element allows for sliding when the horizontal force exceeds the coefficient of friction (μ) times the total gap spring force. Fluid coupling forces are also included in the equations of motion. The fluid coupling forces are assumed to be either in-phase or out of phase with the support motion. The model is excited using simultaneous horizontal and vertical support accelerations.

All the springs have dashpots associated with them to represent structural damping.

This model has been benchmarked against a comparable ANSYS computer model.

7.3 ANSYS TIME HISTORY MODEL DESCRIPTION

To consider the effects of module rocking, sliding and interaction (if it exists), the double rack 3-dimensional ANSYS model is used and is shown on Figure 7.3-1. For illustration purposes this model is a beam representation [Section (1)] of the two racks modeled. However, this model is generated directly from the detailed rack models, described later in Paragraph 7.4 using the super element capabilities of ANSYS.

This time history model is then verified by comparing its fundamental natural frequencies to the detailed finite element model frequencies.

Section No. (2) of this model represents the mass and stiffness of all the fuel assemblies extending the height of the rack. It is pinned at the bottom of the rack and is allowed to impact at the top, top-quarter, and mid-point. A gap at the top-grid, top-quarter, and mid-point represents fuel-to-can clearance. These clearances are represented in the model by ANSYS gap spring elements. Note, also, that this model conservatively assumes that all fuel assemblies are in phase and move together at all times.

The rack is connected to the floor by use of 3-dimensional interface elements. These elements represent two plane surfaces which may maintain or break physical contact and slide relative to each other. At each time step, the program determines if tensile forces exist in the element to see if uplift occurs.

The double rack model includes module interaction or potential for banging with other racks in the pool. Gap springs are located at the top grid elevation and bottom grid and initially have a .25" gap and .50" gap, respectively. This model assumes that the largest interaction occurs for a pair of racks because their rocking motion away from each other is unconfined by lack of adjacent modules.

A single vertical degree of freedom spring represents the pool floor under the racks. The spring rate is calculated based on the floor stiffness and using the mass of both racks.

A structural damping of 4% (DBE) and 2% (OBE) for welded steel frame structures will be used. All internal water entrapped within the rack envelope is added to the horizontal mass. The external hydro-dynamic water mass determination will be based upon a paper by R. J. Fritz entitled "The Effects of Liquids on the Dynamic Motions of Immersed Solids," Journal of Engineering for Industry, February 1972. Both racks for this analysis are assumed to be full of fuel.

The model will account for in-phase fluid coupling with the pool water by use of the ANSYS fluid dynamic coupling element. This element is used to represent dynamic coupling between two points. The coupling is based on the dynamic response of two points connected by a constrained mass of fluid, as described in the aforementioned paper by R. J. Fritz.

An impact damping of 10% will be used for all gap elements. The repetitive impacting of these elements dissipates substantial amounts of energy. Consequently, there is higher damping within the structure than would exist if there were no gaps.

7.4 FINITE ELEMENT MODEL

The computer program ANSYS will be used to analyze the detail finite element model. Deadweight and live load of the fuel will be applied plus the uplift load at the center of the racks.

Figure 7.4-1 delineates the computer model. The spent fuel rack is idealized as a three dimensional detail finite element model of nodal points, consisting of shell and beam elements representing the poison cans and bottom grid.

REFERENCES FOR SECTION 7.0

1. Updated Safety Analysis Report, Fort Calhoun Unit. 1, Appendix F.

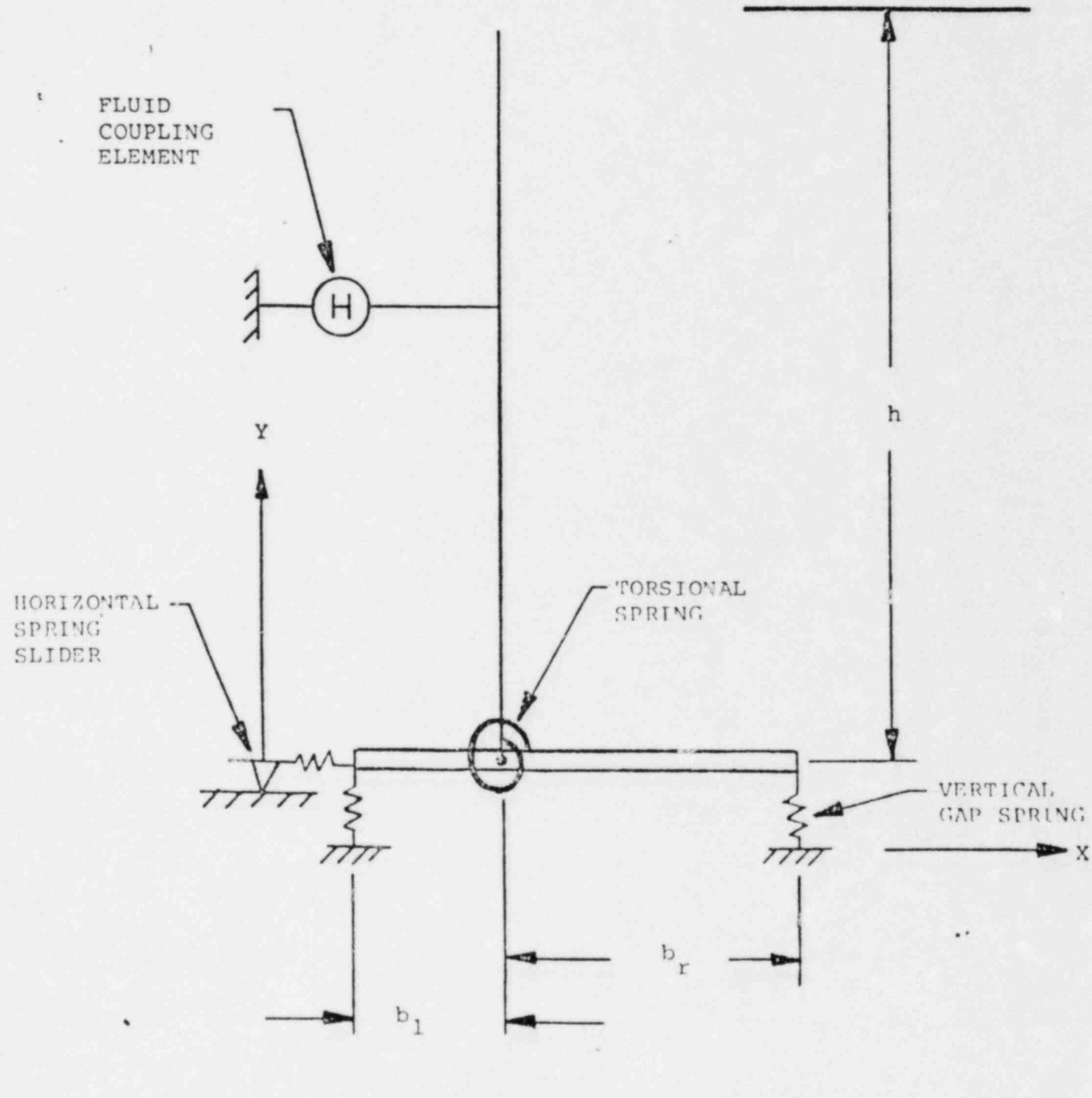
2. ANSYS Rev. 4

3. ANSYS — Computer Program "Engineering Analysis System" Swanson Analysis Systems, Inc.

4. SIMQKE — Computer Program, digitized time histories are generated artifically using "SIMQKE" which was developed under the auspices of the National Science Foundation.

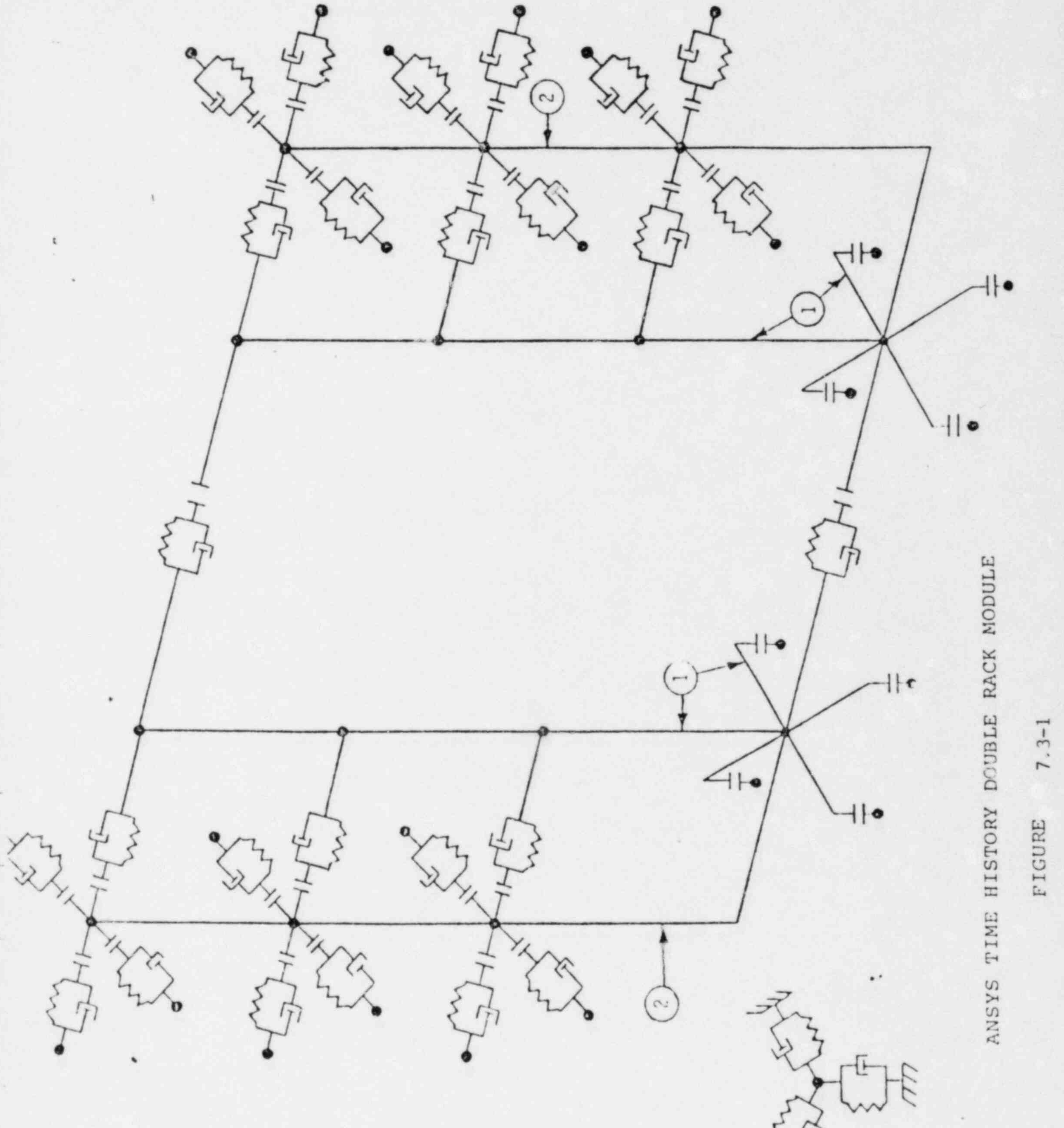
5. RCKN1 — Computer Program for considering nonlinear rocking sliding motion of submerged eccentrically loaded fuel racks. It also considers simultaneous horizontal and vertical history acceleration

6. SPECT — Computer Program, a subroutine of SIMQKE for the computation of spectra from time histories digitized at equal time intervals.



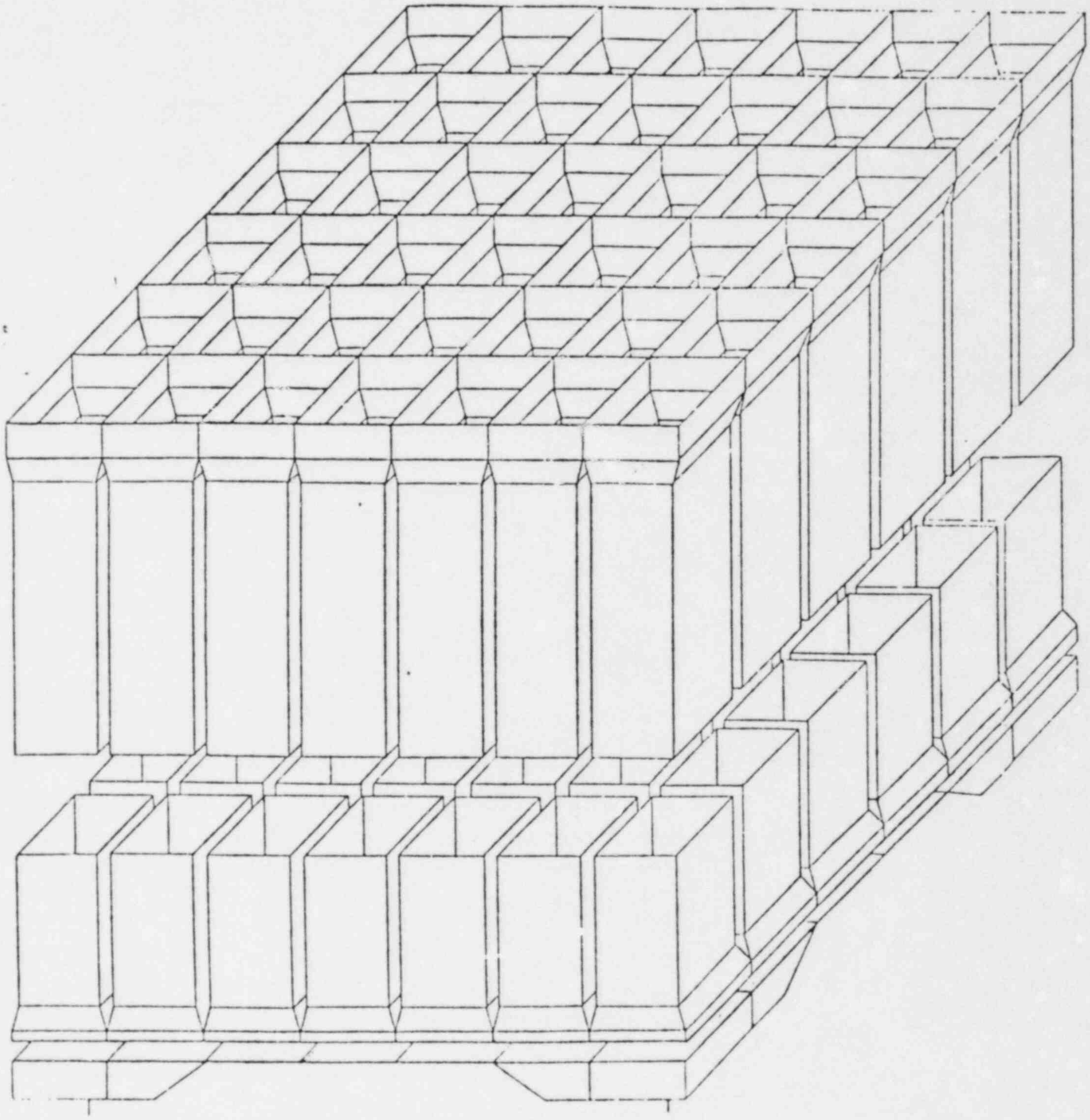
RCKN1 MODEL

FIGURE 7.2-1



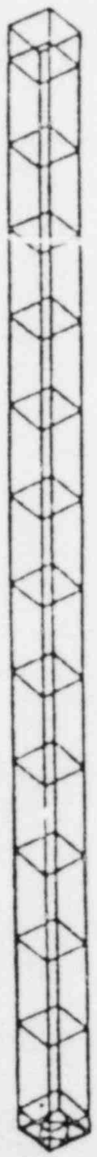
ANSYS TIME HISTORY DOUBLE RACK MODULE

FIGURE 7.3-1



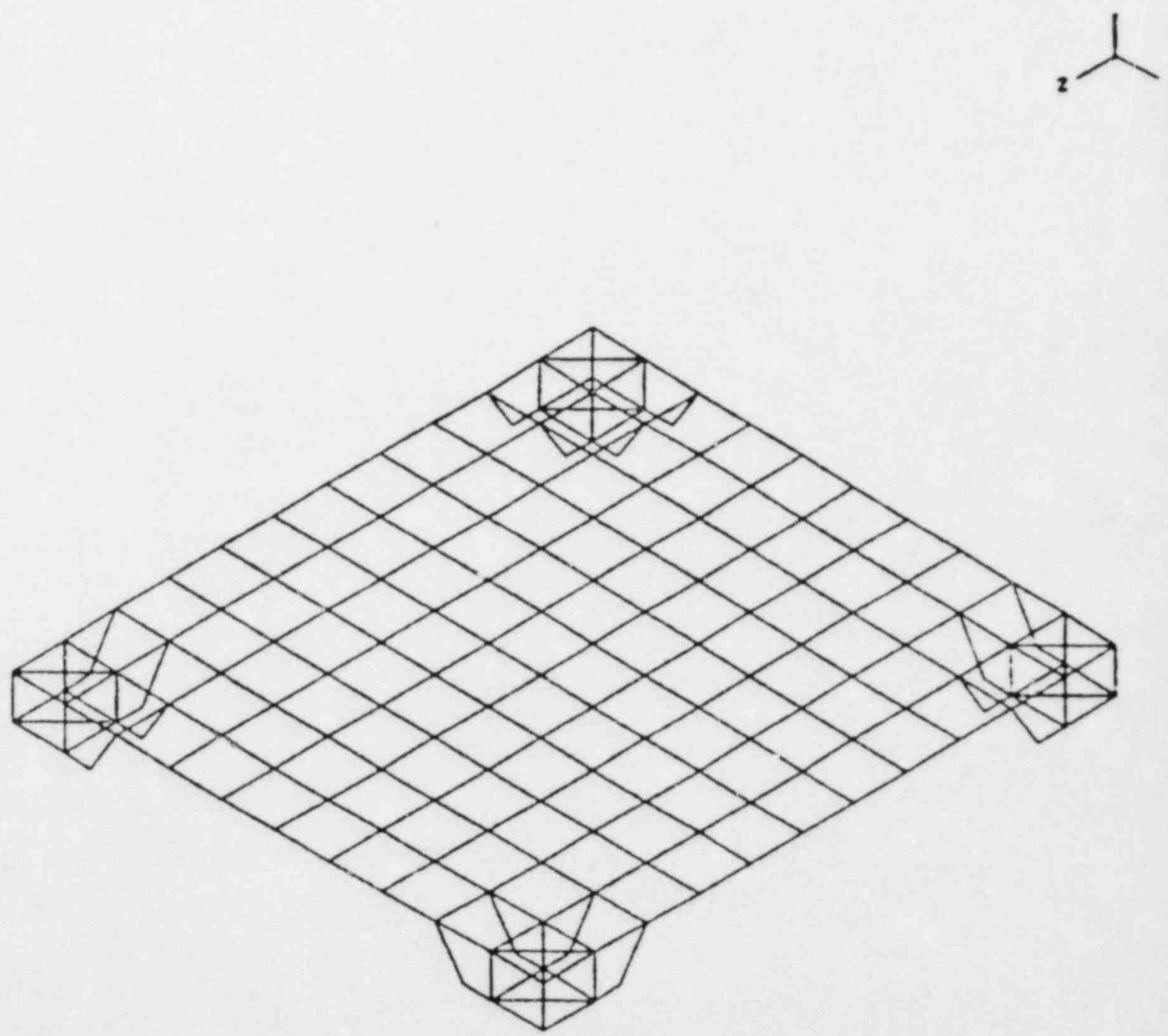
ISOMETRIC VIEW OF FINITE ELEMENT MODEL

FIGURE 7.4-1



CAN GENERATION

FIGURE 7.4-2



[illegible] RACK GRID

FIGURE 7.4-3

8.0 RADIOLOGICAL EVALUATION

The radiological evaluation summarized in this section considers the potential for increases in various radioactivity levels based upon possible changes in fission and activation product inventories associated with the projected increase in spent fuel storage in the spent fuel pool.

8.1 SOLID RADIOACTIVE WASTE

The only solid radioactive waste associated with operation of the spent fuel pool purification system is in the filters and demineralizers, which have solid waste volumes of 5 ft^3 and 45 ft^3, respectively. Materials are usually changed once per fuel cycle, and the projeted increase in spent fuel inventory in the pool will not increase the required frequency of replacement, because the radioactivity which accumulates in the filters and demineralizer is almost entirely the result of corrosion products from the discharged fuel. The radioactivity due to fission products is several orders of magnitude less than that due to corrosion products. Therefore, it is anticipated that the radioactive solid waste generated from the plant will not increase as a result of the increased storage capacity of the spent fuel pool.

8.2 ANNUAL RELEASES OF KR-85

Recent data regarding the annual releases of Kr-85 are shown in the following table:

Annual Kr^{85} Releases (curies)

	Auxiliary Bldg.	Total
1977	53.69	87.79
1978	9.15	23.49
1979	2.93	13.20
1980		6.08
1981		61.46
1982 (Jan.-June)		1.93

The greatest probability for release for Kr^{85} is during the handling of the discharged fuel each cycle. The annual releases are not expected to increase due to an increase of spent fuel stored in the pool.

8.3 POTENTIAL INCREASES IN DOSE TO PERSONNEL

A recent analysis (8/15/80) of the concentrations of the principal radionuclides in the spent fuel pool water provided the following:

Isotope Concentrations (μCi/ml)

Cs-134	3.52×10^{-3}	±	5.53×10^{-5}
Cs-137	7.90×10^{-3}	±	7.50×10^{-5}
Sr-89	0		
SR-90	0		
I-129	0		
I-131	$<2.33 \times 10^{-5}$		
Co-58	5.22×10^{-4}	±	2.25×10^{-5}
Co-60	3.44×10^{-4}	±	2.10×10^{-5}

In the pool, concentrations of 0.01 μCi/ml are typical as shown by the above data. Assuming 1 meV gamma emitters [a reasonable average for Cs-137 (.66 meV), Cs-134 (1.6 meV total, 1.4 meV peak for 27% of the decays), and Co-60 (1.17 and 1.33 meV cascade gammas)], the dose rate due to this contribution is estimated to be 2 mR/hr, in reasonable agreement with doses measured near the spent fuel pool.

Airborne activities in the vicinity of the spent fuel pool have been generally negligible. Samples of airborne activities are only taken when work is scheduled in that area, and concentrations of the principal airborne radiouclides are not available. Significant increases in airborne activity would only be expected to occur during the handling of the fuel discharged each year, and therefore airborne activity is not expected to increase as the result of increasing the total inventory of spent fuel stored in the pool.

There has been no significant increase in potential dose due to the buildup of crud along the sides of the pool. If crud buildup eventually becomes a major contributor to pool doses, measures will be taken to reduce such doses to ALARA.

The total doses for operations personnel in the fuel pool area during 1980 were as follows:

Refueling Operations	290 mrem
Maintenance, testing and preoperation of spent fuel handling machine	28 mrem
Total	318 mrem

During normal refueling operations the freshly discharged 1/3 core will be placed in the poisoned region (Region 1).

The fuel will then be measured for fuel depletion and be moved to Region 2 after fuel burn-up has been documented. Thus the fuel will be moved twice after each refueling.

It can be expected that the dose rates would be slightly higher with increased fuel handling. For a very conservative evaluation the total dose rates listed above can be doubled since the fuel is moved twice. This would result in a dose rate of 640 mrem per refueling which is still under acceptable limits. This is a very conservative dose rate, since it does not account for any decay time between discharging the fuel from Region 1 to Region 2, (burn-up calculations would be done prior to moving fuel to Region 2).

Based on the information previously described in this section, a best estimate evaluation indicates there will be no significant increase in dose to plant personnel, or at the site boundary, as a result of the proposed modification. In addition to this best estimate evaluation, a conservative evaluation has been made in which it was assumed that all radioactivity levels are proportional to the relative inventory of fission products contained in the spent fuel stored in the pool. Even with this conservative assumption, the maximum radiological impact of the proposed modification is clearly acceptable.

Based on total fission product inventories, the maximum possible increase in the peak dose rates following a normal refueling is only 20% greater than those currently experienced. In addition, based on the same logic, the maximum possible increase in the average annual dose which could only occur in the final year of plant operation would be predicted to be only 80%. However, plant data supports the conclusion that significant fission product activity is only released during refueling, and therefore the maximum possible increase in average annual dose is also 20%.

Finally, it should be noted that the dose impact of normal spent fuel storage to operating personnel and at the site boundary are small compared to those resulting from other sources of radioactivity associated with the operation of a nuclear power plant. Therefore, even a conservative overestimate of the maximum possible radiological impact of the proposed modification indicates that only small increases in already small doses would result from the proposed modification. As such, the small increases are acceptable.

8.4 RACK INSTALLATION

The pool modification consists only of replacing, in a predetermined sequence, the presently installed spent fuel storage racks with the new high density racks by simply lifting out the present ones and lowering the new ones in place.

Spent fuel assemblies must be moved during this operation in order to sequentially empty the rows of racks to be removed and store the fuel in the rows of new racks after installation. Although the total inventory of stored spent fuel assemblies will have to be moved, only approximately 28 of these assemblies will be fuel that was discharged since the last refueling outage. The total accumulated dose during these fuel handling operations is expected to be of the same order as that experienced during a normal refueling. Based on the 1980 refueling outage, the total dose for operations in the fuel pool area was .32 man-rem, and therefore, the dose due to fuel handling during rack installation would be expected to be less than 1 man-rem.

Any significant increase in personnel dosage due to this modification beyond that incurred during regular refueling outages would be from the rack decontamination process. A conservative estimate is that any increase will be considerably less than that experienced during the changeout of the SFP filter. Using one-third of the dose estimate for filter changeout of .6 man-rem for 14 hours elapsed time, results in a dose rate of .015 man-rem/man-hour. Assuming 6 man-hours to decontaminate one rack, the total dose for 21 racks would be 1.89 man-rem.

Adding this estimate to the dose due to fuel handling operations (less than 1 man-rem) results in a total dose for the entire pool modification of less than 3 man-rem.

Below is an outline of the actions that will be taken to assure that occupational doses during each task of the pool modification will be ALARA:

1. A health physicist will be present at all times during the re-racking to monitor for excessive airborne or high radiation by utilizing portable or hand-held radiation monitoring instruments,
2. Area Radiation Monitors will be used to alarm on a high radiation signal,
3. Personnel shall be required to wear appropriate clothing as determined by the health physicist to preclude contamination,
4. As the racks are pulled out of the water, they will be washed,
5. All rack decontamination areas will be enclosed by a cover-all to reduce airborne contamination,

6. The District is considering the possibility of also deconning the racks in the pool by using chelating agents, along with agents to dissolve surface films and remove surface crud. None of the chemicals which will be added to the pool will initiate stress corrosion cracking or change the pH significantly, and
7. All personnel will be required to undergo 2 days of radiation training.

The spent fuel pool area is monitored by an area monitoring system. Actual dose rates can be read locally and in the control room. The monitor will alert personnel in the SFP area when dose rates exceed the setpoints. This monitor, RMO-85, is located outside the SFP wall as shown on Figure 8.4-1. Health Physicists carrying portable monitoring instruments will also provide a further means to identify increases in radiation levels. Additionally, a radiation monitor is located on the spent fuel handling machine to indicate dose rates in this area.

All effluents from this area are discharged through the auxiliary building stack, via the auxiliary building HVAC system and are monitored through gaseous effluent and particulate monitors RMO 60/61/62. In the event that these gaseous effluent monitors indicate activity levels are in excess of the alarm limits, the auxiliary building flow paths are manually closed. The exhaust ventilation ductwork from the spent fuel storage area is equipped with iodine absorbers which are manually brought into operation whenever irradiated fuel is being handled.

The space between the spent fuel pool racks and inside concrete shield wall will not be reduced due to this modification. In fact, this distance will be increased at all locations and accordingly, the dose rate outside the concrete wall is not expected to increase beyond present levels.

8.5 RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS

Loss of Normal Spent Fuel Pool Cooling

The spent fuel pool cooling system has been reviewed entirely to determine if any single active failures could preclude the system's cooling capabilities. It was found that there is no single active failure which could preclude spent fuel pool cooling. However, in the unlikely event that a failure would occur it has been determined that the pool could be cooled indefinitely by recirculating and adding water from the SIRW tank and Fire Hoses. The worst case accident would be to have a failure occur immediately after a refueling outage while the plant is at full power. The heat load generated from this fuel off load would be 5 x 10^6 BTU/hr. (assuming an eight week decay). If an accident were to occur to prevent spent fuel pool cooling, the pool would heat up approximately 2.885°F/hr., and it would take 32 hours to reach the pool's boiling point without any supplemental addition of water from the SIRWT or Fire Hoses. Also, as discussed in Section 5.2, alternate/supplemental cooling can be provided by the shutdown cooling system, should a loss of spent fuel pool cooling system occur during a full core discharge. However, the plant cannot use the shutdown cooling system to supplement the spent fuel pool cooling system while at full power (See Sect. 5.0). Even in the event that the normal spent fuel pool cooling system is inoperable, and supplemental water is not able to maintain the pool temperature below boiling, this improbable incident does not result in serious consequences since the pool can withstand boiling conditions. The radiological consequences of a boiling pool were analyzed and found to be acceptable as described in the following paragraphs.

As discussed above, the pool walls, the liner, and the fuel assemblies are designed to withstand boiling temperatures without a loss of integrity. The radiological consequences from pool boil off would therefore be limited to release of radionuclides normally present in the pool water. Maximum concentration of radionuclides which may be present in the pool are listed in Section 8.3. A volatile Isotope such as I-131, is the only Isotope which may be available to be released in significant quantities in the event of a pool boil off, since non-volatile Isotopes have a very high decontamination factor. The concentration of I-131 activity available for release would be: 2.33×10^{-5} µCi/cc times the total pool volume (7.36×10^{8}cc) which equals 1.17×10^{4} µCi or 1.17×10^{-2} Ci. This release would be insignificant compared to 1.93×10^{4} Ci of I-131, 7.5×10^{2} Ci of Kr-85, 1.57×10^{2} Ci of Xe-131m, and 3.89×10^{4} Ci of Xe-133 which are available for release during a Fuel Handling accident (Reference: Section 14.18.4.2 of the USAR). The above comparison is made since it has already been concluded in the reference section of the USAR, that the radiological consequences of a Fuel Handling accident in the spent fuel pool is within 25% of 10 CFR 100 limits. Therefore, it can also be concluded that the radiological consequences, in the unlikely event of a pool boil off will be well within 25% of 10 CFR 100 limits, since the activities available for release are significantly lower than those postulated in the event of a fuel handling accident.

Radiological Effects of a Dropped Fuel Canister

Various drop accident analyses were performed which address the effects of dropping a fuel canister into the spent fuel pool. Postulated drop accidents include: 1) a straight drop on top of the rack, 2) a straight drop through an individual cell all the way to the bottom of the rack, 3) an inclined drop on the top of the rack, and 4) one fuel canister impacting another fuel canister already in a storage cell.

The radiological releases from these various cases were found to be well within 25% of 10 CFR 100 limits. A direct impact on the fuel pool liner would also envelope the radiological consequences of this accident and are within 25% of 10 CFR 100. The consequences due to this postulated load drop would be the same as dropping a fuel assembly directly onto the spent fuel pool floor which has been previously analyzed as part of the USAR. These cases assume impacts of the highest burned up fuel rods, 72 hours after shutdown.

Since there will be a negligible change in radiological conditions due to the increased storage capacity of the spent fuel pool, no change is anticipated in the radiation protection program. In addition, the environmental consequences of a fuel handling accident in the spent fuel pool, described in the USAR, Section 14.18 remains unchanged. Therefore, there will be no change or impact to any previous determinations of OPPD Final Environmental Statements.

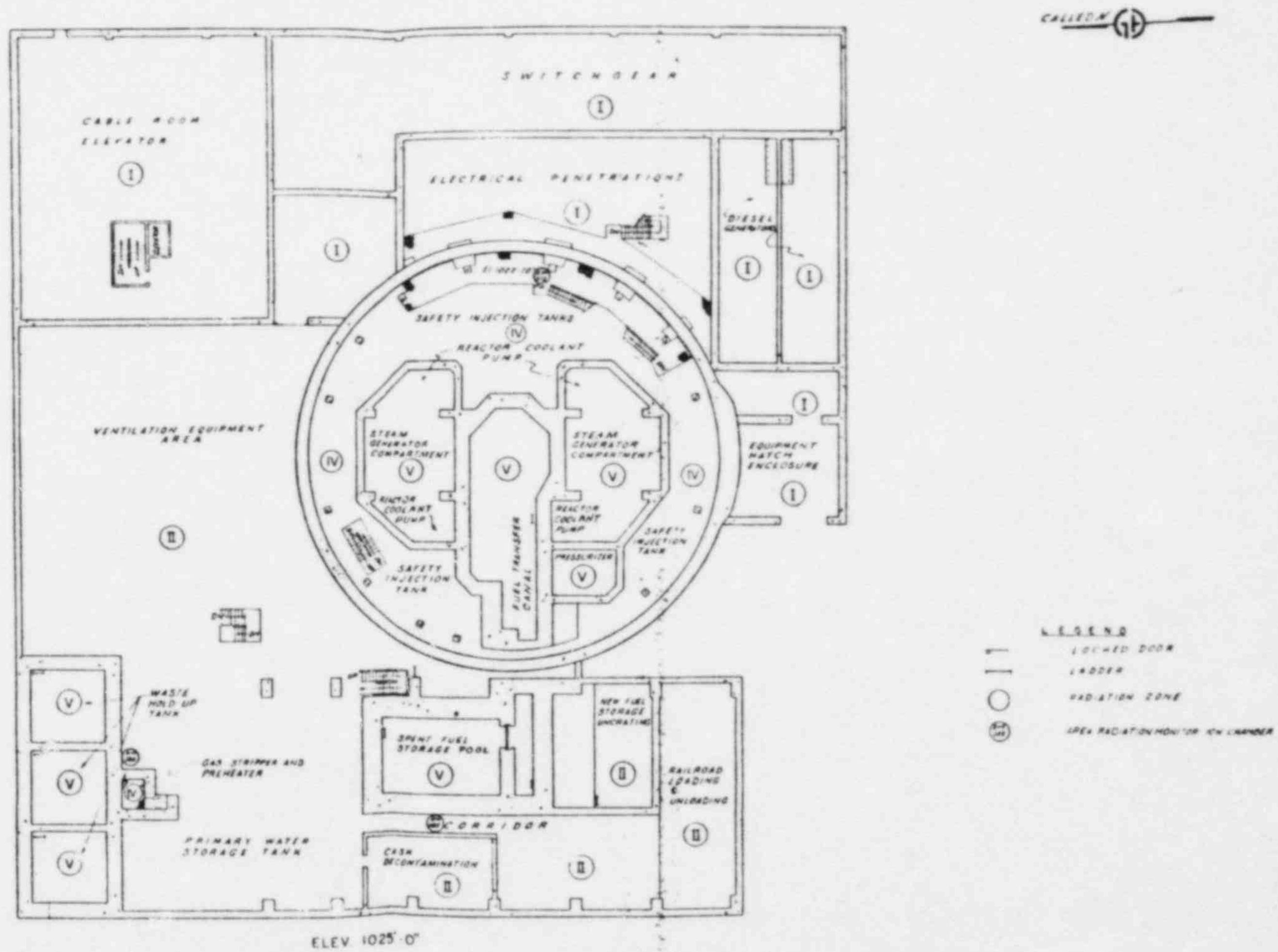
8.6 RADIOLOGICAL CONSEQUENCES DUE TO A SEISMIC EVENT IN THE SPENT FUEL POOL

It will be verified that the seismic loads imposed on the fuel pool liner walls would not result in any damage to the liner such as to cause 1) significant releases of radioactivity due to mechanical damage to the fuel, 2) significant loss of water from the pool which could uncover the fuel and lead to release of radioactivity due to heatup, 3) loss of ability to cool the fuel due to flow blockage caused by a portion or one complete section of the liner plate falling on top of the fuel rack, 4) damage to safety related equipment as the result of pool leakage and 5) uncontrolled release of significant quantities of radioactive fluids to the environs.

The SFP liner is designed to support all postulated or effective dead and live loads, hydrostatic loads, temperature gradients to 212°F, and the effects of a design basis earthquake (DBE). Drainage grooves are provided behind the stainless steel liner which permit detection of any liner leakage. As per Section 5 and 9.5 of the USAR, the liner is designed to withstand these seismic loads without failing, and thus, no postulated accident could damage, uncover the fuel or result in radioactive releases in excess of the 10 CFR 100 limits.

The new spent fuel storage racks have been designed not to exceed the spent fuel pool liner plate allowable unit floor loading of 500 psi. Analysis will verify that the unit floor loading imposed by all dead weights, together with the combined vertical components of seismic accelerations as determined by the seismic analyses, will not exceed the 500 psi limit.

Thus, the seismic loads imposed on the fuel pool liner walls will not result in damage to the liner and there would be no consequential releases of radioactivity from the pool liner due to mechanical damage to the fuel. There would be no significant losses of water from the pool which could uncover the fuel, since the liner is designed to withstand a DBE. No safety related equipment is within the postulated leakage area. The uncontrolled releases of significant quantities of radioactive fluids to the environs would not occur, since the liner plate is designed to prevent leakage during the most severe accidents.



OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION UNIT NO. 1
WORK ORDER NO. 1001

INTERMEDIATE FLOOR
RADIATION ZONES

Gibbs & Hill, Inc.
ENGINEERS DESIGNERS CONSTRUCTORS
NEW YORK
JOB NO.

SCALE 1/8" = 1'-0"

FIG. 8.4-1

9.0 ENVIRONMENTAL IMPACT EVALUATION

In August 1979, the NRC issued the Final Generic Environmental Impact Statement on Spent Fuel Storage which indicated that interim storage of spent fuel at individual nuclear power stations was acceptable. The statement concluded that:

1) Storage of light-water reactor fuel in water pools has an insignificant impact on the environment.
2) Modificiations of on-site spent fuel storage pools to increase their capacity are environmentally acceptable.

Earlier, OPPD came to these same conclusions as described in its April 19, 1976 licensing submittal which requested approval to install the present racks. The following list summarizes the basic conclusions reported in that earlier submittal which are equally valid for the proposed racks:

1) It is not likely that the modification would constitute a commitment of resources that would affect the alternatives available to other nuclear power plants or future actions taken by the industry to alleviate fuel storage problems.

2) The proposed modificiations will require custom made racks made of stainless steel and Boraflex®. These materials are readily available and are in abundant supply. The total material requirement is insignificant and does not present an irreversible commitment of natural resources.

3) There are no potential effects on the environs outside of the fuel storage building that will result from the proposed construction work. Within this building, the impacts are expected to be limited to those normally associated with metal-working activities.

4) There are no adverse effects that will occur on-site or in the surrounding environs that can be associated with an increase in the number of fuel assemblies stored in the pool.

These conclusions are further supported by the radiological evaluation summarized earlier in Section 8.0 of this document.

10.0 COST-BENEFIT STATEMENT

Section 2.0 of this document summarized the evaluation conducted by OPPD of alternatives for disposition of spent fuel to be discharged from Fort Calhoun. That evaluation determined that several of the alternatives did not represent viable solutions to the problem, and therefore, the cost of those alternatives was immaterial since there was little or no benefit. Reprocessing, fuel cycle optimization, shipment to another site, and storage in a independent spent fuel storage facility owned by someone other than OPPD are all alternatives which are currently not viable.

A viable alternative would be the construction, either on or off site, of an independent spent fuel storage facility by OPPD. Although there are considerable potential difficulties in siting such a facility, the major disadvantage of this alternative is cost. Cost estimates of such a facility vary widely, but a range of $125 to $150 per kg U is not unreasonable. In 1981 dollars, this estimate corresponds to 37.8 to 45.3 million dollars to store the excess fuel to be discharged from Fort Calhoun.

Use of portable casks to store fuel on site also appears to be a viable alternative assuming such casks are available. There is also a large uncertainty associated with the estimated costs of this alternative but a range of \$75 to \$150 per kg U seems reasonable. This corresponds to a total cost of between 22.7 and 45.3 million dollars. Use of non-portable casks or other above ground structures for dry storage appears to have the potential of achieving costs in the range of \$50 to \$75 per kg U, which corresponds to a total cost of 15.1 to 30.2 million dollars. However, these concepts have not yet been licensed in the United States, and both the licensing and production schedules for these concepts are currently subject to large uncertainties.

Use of subsurface, on-site storage concepts may be feasible, but costs are estimated to be between \$100 to \$150 per kg U, which corresponds to a total cost of 30.1 to 45.3 million dollars.

An alternative not discussed in Section 2.0 is the shutdown of the Fort Calhoun nuclear plant when the present spent fuel storage space is exhausted. If the existing racks are completely filled with discharged fuel and it doesn't become necessary to off load the entire core between 1985 and 1990, then operation could continue until the 1990 outage. A conservative estimate of the cost of a premature shutdown was obtained by considering only the differential of fuel costs between Fort Calhoun and the lowest cost fossil unit available to OPPD. Note that this cost estimate neglects any costs resulting from the unamortized capital costs of the facility, and assumes replacement energy capability is available. The cost to OPPD of shutting down Fort Calhoun in 1990, instead of 2008 is 524 million dollars.

Finally, costs were estimated for both the new spent fuel storage racks and the disassembly and compact storage of all fuel necessary for operation through the year 2008. This total cost is estimated to be 3.9 million dollars or $12.9/kg U. The cost of the new racks is slightly less than one-half of this estimated cost and is a reasonably well known cost based on past experience. Although the costs of disassembly and compact storage may be subject to a large uncertainty, doubling the cost estimate of such operations would result in a cost estimate of only $20/kg U. Therefore, it is clear that the selected alternative for disposition of spent fuel from Fort Calhoun is the least cost alternative by a large margin. In addition, implementation of this alternative will allow the customers of OPPD to continue to benefit from the low cost power being generated by Fort Calhoun. Finally, the selected alternative provides the flexibility to respond to changes in government or industry programs by providing for fuel storage in the conventional manner until 1994.

As discussed earlier in this document, this facility modification will not have any significant impact on the environment, or the public health and safety. In view of this and the obvious cost benefits described above, it is clear that the proposed modification is the most prudent and cost effective alternative available.

11.0 QUALITY ASSURANCE, TESTING, AND IN-SERVICE INSPECTION

11.1 Material, Quality Control, Testing, and In-service Inspection

The materials used in the fabrication of the spent fuel storage racks are per the table listed in Section 3.0. Certified Material Test Reports will be required, including actual chemical and physical test results, for materials used in the rack fabrication. The weld filler materials also will have, as a minimum, certificates of compliance and ferrite data.

The fabrication of the racks will be in accordance with OPPD contract documents which includes the following QA/QC procedures:

a. Manufacturing and inspection plan
b. Cleaning procedures
c. Liquid penetrant inspection procedures
d. Visual weld inspection procedures
e. Welding procedures
f. Packaging, shipping, and handling procedures
g. Final inspection/test procedures
h. Weld repair procedures
i. QA documentation checklist
j. Fabrication and inspection procedures

11.2 Poison Verification Program

An on-site test will be performed to confirm the presence and retention of the neutron absorber in the racks. The presence of Boraflex® will be completely documented by Quality Assurance personnel during construction. Final verification will be performed by a neutron attenuation test on a minimum of 15% of the cells in each storage rack, which contains poison, selected at random. A neutron source and applicable detectors will be used to verify that the four sheets of Boraflex® are in their correct position adjacent to the cell walls. Should this test program identify the absence of even one Boraflex® sheet, then a 100% testing of the cells will be performed. Verification of the retention of the neutron absorber will be accomplished by an In-Service Surveillance Test detailed in the next section.

11.3 Test and In-Service Inspection

Poison test coupons are supplied for an in-service program of the neutron absorber used in the high density fuel storage racks. The coupons will be located between two fuel cells, and will thus receive the highest attenuation, since two of the most recently discharged fuel assemblies with the highest burnup rate will be placed in those cells. The coupons duplicate the condition of the Boraflex®, which is encased in the poison cells. Ten coupons will be provided, with three samples from each coupon removed and analyzed at intervals of 1, 2, 4, 7, 11, 15, 20, 25, 30, and 35 years after installation.

The coupons will be tested along the following guidelines:

1. Boraflex® samples are cut to size and conditioned in normal atmosphere at 20°C to 30°C and 30 percent to 70 percent R. H. for 3 days.

2. Each sample has the following taken at predetermined points on the samples.
   a. Dimensions, thickness, length and width
   b. Hardness on the Shore A scale
   c. Neutron Attenuation at .06eV

3. Each sample is then fabricated into a coupon and this coupon is installed in the pool.

4. A coupon is to be removed per the above schedule.

5. The coupon is carefully opened without damaging the sample and is conditioned in accordance with Step 1.

6. The samples are remeasured as indicated in Step 2, insuring that measurements are made in the same location and by the same procedure.

7. The surface is visually examined and photographed.

8. Reports of the sample test are documented.

9. If adverse conditions are noted, then the samples may be subjected to further neutron transmission tests.

10. The samples are retained.

A reference neutron absorber sample will be provided for each coupon. This reference sample is from the same strip of neutron absorber as the coupon sample. This reference sample can be used for future comparison to the installed sheets in the test coupons.

APPENDIX A

DRAFT

SAFE LOAD PATH FOR REMOVAL AND INSTALLATION

OF SPENT FUEL POOL RACKS

(Preliminary)

1. INTRODUCTION

The District is planning to replace the exisiting spent fuel pool racks with high density racks in 1983. The existing racks will be removed from the spent fuel pool, decontaminated and shipped off-site. New high density racks will be moved from the railroad bay area to spent fuel pool for reracking. The existing racks are 49.625" x 90.125" x 168.5" high and weigh approximately 9,600 pounds. New racks are 80.23" x 98.165" x 168.5" high and weigh approximately 15,800 pounds. These loads are greater than the combined weight of a single spent fuel assembly and its handling tool. (approximately 1,500 pounds) and therefore, fall under the category of heavy loads as defined in the NUREG-0612, "Control of Heavy loads at Nuclear Power Plants." A procedure meeting the intent of NUREG-0612 was developed.

II. ASSUMPTIONS

The load path developed is based on the following assumptions:

1. Racks will be lifted or lowered in the southeast part of the railroad bay area. (See Figure A-1)

2. The racks will not be carried over the in-place new or existing racks containing fuel, in the spent fuel pool.

3. The spent fuel storage gate is closed and the fuel transfer canal drained, while racks are being moved in the area.

4. The lifting mechanism and its lugs are designed in accordance with the requirements of ANSI 14.6, 1978, "Standards for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials".

5. The SFP racks will be decontaminated in the railroad bay area at elevation 1004'-0" and shipped off-site.

6. The auxiliary building crane will be operated over the spent fuel pool in accordance with Fort Calhoun Technical Specifications 2.11 and Operating Instruction OI-HE-5.

7. Rack removal sequence is per Section 3.5 and paragraph 3 of this document.

III. PREREQUISITIES

1. A piece of hand rail at 'B' location should be cut to provide a space wide enough to allow passage of the racks. These pieces of handrail should be welded back after the reracking.

2. The auxiliary building crane should be inspected, tested and maintained in accordance with the applicable requirements of ANSI B30.2-1976, "Overhead and Gantry Cranes".

3. Proposed load paths have been discussed with the crane operator and signal man and a dedicated signal man is available to guide the crane operator.

IV CRITERIA FOR SELECTION OF THE LOAD PATH

Criteria for selection of the load path are as follows:

1. No safety related equipment or irradiated fuel is impacted by the dropped load along the load path.
2. the load path will follow structural floor members, beams, etc., to extent practical. In the event of a load drop, and structural member failure, no safety-related equipment on the intermediate floors or basement floors will be damaged.

V. DESCRIPTION OF PROPOSED LOAD PATH

The proposed load path is shown on the attached figure A-1. If any deviations are planned to be made from the proposed path, they will first be reviewed, and approved by the Plant Review Committee prior to moving the racks.

Removal of Existing Racks

The existing SFP racks will be removed from the spent fuel pool in a sequence as outlined in Section 3.5 paragraph 3. The racks will be moved from the spent fuel pool to the railroad bay area in the following manner.

1. The racks will be lifted out of the spent fuel pool and moved directly towards the south wall of the spent fuel pool. From this position, the SFP racks will be brought directly over the fuel transfer canal and move east over the transfer canal to location 'C', with the bottom of the SFP racks at elevation 1039'-6". By stopping here, the Safety Injection and Refueling Water Tank (SIWRT) will be cleared and the rack will not interfere with the fuel handling machine distance gap.

2. The SFP racks now will be turned south and moved to position 'B'. At this position, the hand rail may have to be cut to allow the rack to pass. From position 'B' the SFP racks should be moved south to location 'A', the centerline of beam B-64 located at elevation 1004'-0" and lowered into the railroad bay area. The SFP racks should be decontaminated in this area and shipped off-site.

Installation of New Racks

The new racks should be installed in the reverse sequence as described in the removal procedure i.e. lifted at position 'A' on the railroad bay area to an elevation of 1039'-6" and follow the described load path to the spent fuel pool.

VI. AREAS AFFECTED BY PROPOSED LOAD PATH

The following areas will be affected by the proposed load path.

a. Railroad Bay Area - The southesast area of the railroad bay will be affected by this load path. There is no safety-related equipment beneath this structure. The safety injection and refueling water tank is not endangered as the load path avoids that.

b. New Fuel Storage and Uncrating Area - The loads in this area travel over a series of different slabs and beams at different elevations. A load drop along the proposed path in this area could conceivably fail the various intermediate slabs and fall to the basement floor (elevation 989'-0"). However, since there is no safety-related equipment in this area, this fall would not cause a plant safety problem.

c. Fuel Transfer Canal - A load drop in the fuel transfer canal would not damage spent fuel, since no spent fuel is stored in the canal. The bottom of the transfer canal consists of a stainless steel liner plate over 12-foot thick concrete mat on caissons. The worst postulated damage would be the puncturing of the stainless steel liner plate but the concrete behind the plate would retain its integrity.

d. Walls Around Spent Fuel Pool - The walls around the spent fuel pool at elevation 1039'-0", are approximately 4-foot thick. A SFP rack drop may cause a local spalling of the concrete but will not cause any significant damage.

e. Spent Fuel Pool - The racks in the spent fuel pool will be moved in accordance with the sequence listed in Section 3.5 paragraph 3. This procedure will ensure that the racks do not travel over the in-place new or existing racks containing fuel, in the spent fuel pool.

The travel of both the main hook and the auxiliary hook is prevented over the spent fuel pool by electrical interlocks. Fort Calhoun Technical Specification 2.11 and Operating Instruction OI-HE-5 state that the Auxiliary Building Crane should not be used to move any loads in the spent fuel pool without a written procedure. This procedure will describe all the precautions and conditions and be approved by the Plant Review Committee. A key to bypass the electrical interlocks can be obtained from the Shift Supervisor after the procedures are complete. Operating Instructions OI-HE-5 requires that a certified/qualified crane supervisor be in administrative control when the electrical interlocks are inoperative. The interlocks will be required to be bypassed in order to remove the racks out of the pool area.

The reracking sequence procedures and administrative control of the crane would preclude any possibility of a load drop over the stored fuel resulting in the radiological release.

VII. SAFETY EVALUATION

The SFP racks handling does not constitute an unreviewed safety question (per 10 CFR 50.59) for the following reasons.

1. The probability of an occurence or the consequences of an accident is not increased provided that the rack follows the prescribed path. A load drop will not damage any safety equpment.

2. The administrative control and reracking sequence procedures, would preclude the possibility of a load drop on any safety related equipment.

3. Since the SFP racks line of travel does not effect any safety equipment, the margin of safety, as defined in the basis for any technical specification is unaffected.

VIII. CONCLUSION

The load path described above for the removal and installation of racks is safe and meets the intent of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

IX. REFERENCES

1. "Sample Cask Drop Analysis and Removal Procedures", at Fort Calhoun Station by Stone and Webster Engineering Corporation, dated June 7, 1978.

2. "Control of Heavy Loads at Nuclear Power Plants, Reponse prepared by Stone and Webster Engineering Corporation for Omaha Public Power District".

3. Gibbs Hill Durham and Richardson Drawings.

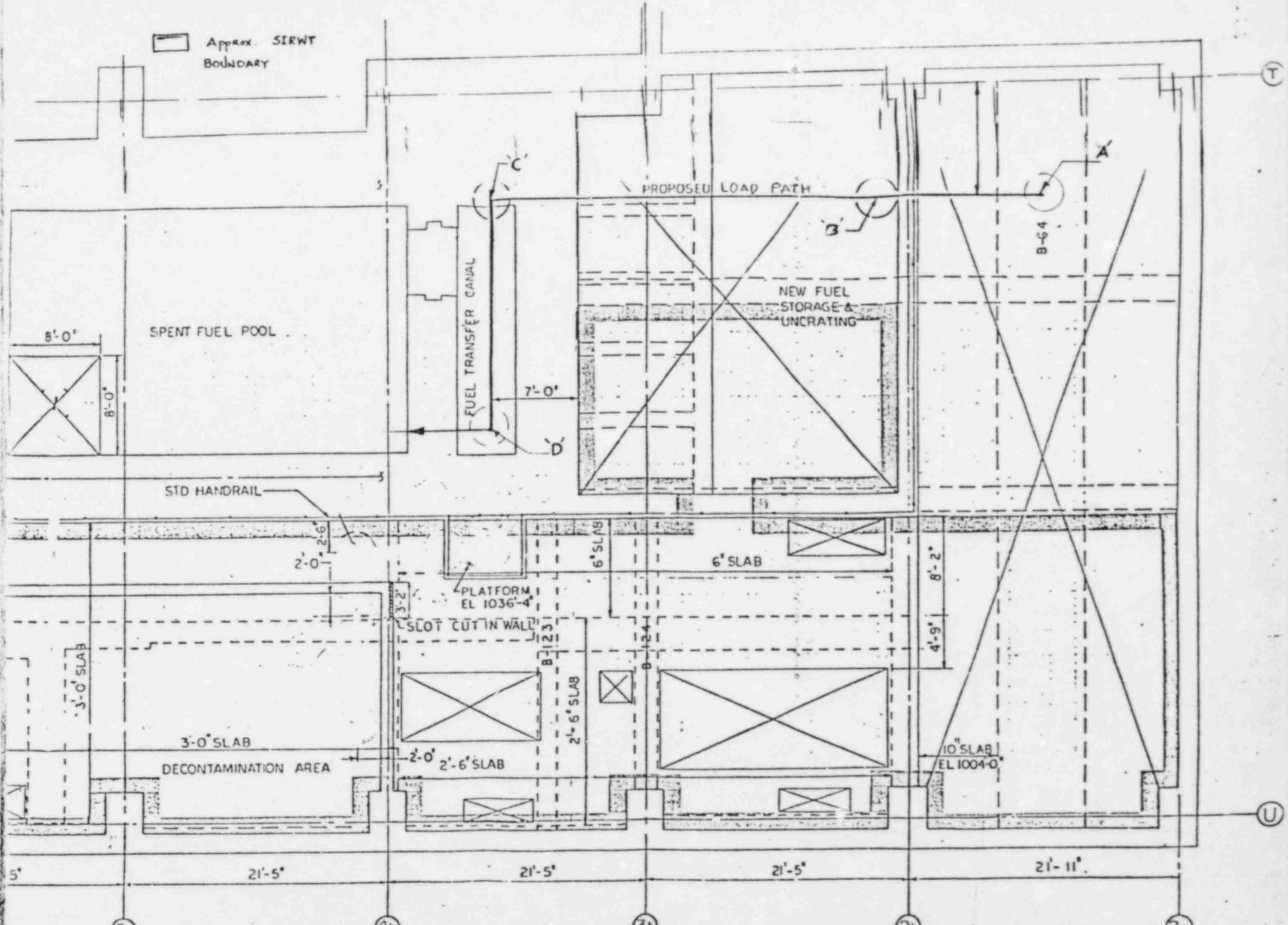


FIGURE A-1

PROPOSED LOAD PATH FOR REPACKING OF SFP RACKS