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DESCRIPTION:

Ltr on behalf of Rochester Gas & Elec Co...... w/attch certificate of service...notarized... 1-30-76....trans the following:

PLANT NAME: Ginna

ENCLOSURES:

Amdt to OL/Change to Tech Specs: Consisting of revision to tech specs with regard to modification to spent fuel pool storage racks....(40 cys encl rec'd)

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January 30, 1976

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RECEIVED JAN 30 1976 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

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

Re: Rochester Gas and Electric Corporation
R. E. Ginna Nuclear Power Plant Unit No. 1
Docket No. 50-244

Dear Mr. Rusche:

As counsel for Rochester Gas and Electric Corporation, we hereby transmit three (3) signed originals and nineteen (19) copies of a document entitled "Application for Amendment to Operating License" together with forty (40) copies of a proposed change to Technical Specification 3.8.1. This request for change in technical specifications is being submitted in connection with proposed modifications to the spent fuel pool storage racks for the Ginna plant. Attachment B to this application sets forth the safety evaluation for the proposed change in specification as well as a complete description of the proposed modifications.

In the opinion of RG&E's Nuclear Safety and Audit Review Board, the proposed modifications do not constitute an unreviewed safety question within the meaning of 10 C.F.R. § 50.59(a), as discussed more fully in Attachment B. Since, however, it has been the Commission's recent practice to review and, in effect, approve all modifications dealing with spent fuel storage pools, RG&E hereby requests approval of these modifications in addition to approval of the proposed change in specifications.

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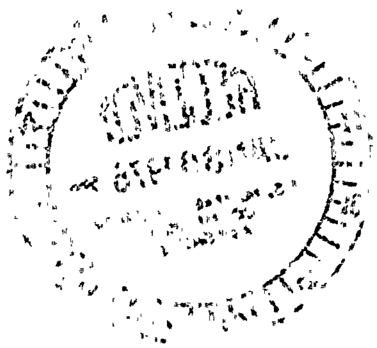

Forty (40) copies of Attachment B are also enclosed.

A Certificate of Service showing service of these documents upon the persons listed therein is also enclosed.

Very truly yours,

LeBoeuf, Lamb, Leiby & MacRae

LeBoeuf, Lamb, Leiby & MacRae
Attorneys for Rochester Gas
and Electric Corporation

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Rochester Gas and Electric Corporation	)	Docket No. 50-244
(R. E. Ginna Nuclear Power Plant, Unit No. 1)	)	

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the U.S. Nuclear Regulatory Commission (the "Commission"), Rochester Gas and Electric Corporation ("RG&E"), holder of Provisional Operating License No. DPR-18, hereby requests that Technical Specification 3.8.1 set forth in Appendix A to that license be amended. This request for a change in the technical specifications is submitted in view of proposed modifications to the spent fuel pool storage racks which will increase the storage capability of the pool.

The proposed technical specification change is set forth in Attachment A to this Application. A safety evaluation demonstrating that the proposed change does not involve a significant change in the types or a significant increase in the amounts of effluents or any change in the authorized power level is set forth in Attachment B. Attachment B also describes the proposed modifications to the spent fuel pool storage racks and supports

the conclusion by Applicant's Nuclear Safety Audit and Review Board that the modifications to the facility as described in the facility's Technical Supplement Accompanying Application for a Full-Term Operating License do not constitute an unreviewed safety question within the meaning of 10 C.F.R. § 50.59(a) of the Commission's regulations.

WHEREFORE, Applicant respectfully requests that Appendix A to Provisional Operating License No. DPR-18 be amended in the form attached hereto as Attachment A.

Rochester Gas and Electric Corporation

By L D White Jr.
L. D. White, Jr.
Vice President
Electric and Steam Production

Subscribed and sworn to before me this 26th day of January, 1976.

Walter C. Hildebrandt
WALTER C. HILDEBRANDT
NOTARY PUBLIC, State of N. Y., Monroe County
My Commission Expires March 30, 1977

ATTACHMENT A

Add paragraph 3.8.1,g and h to Section 3.8.1.

3.8.1,g. The decay heat of the fuel stored in the spent fuel pit plus the fuel removed from the reactor for normal refueling will not exceed 5.3×10^6 BTU/hr.

3.8.1,h. If the full core is to be placed in the spent fuel pit, the decay heat of the fuel stored in the spent fuel pit plus the fuel removed from the reactor will not exceed 9.3×10^6 BTU/hr.

Add the following to the end of the Basis for Section 3.8.1.

During normal refueling the spent fuel pit temperature is limited to 120°F[3]. At this temperature the spent fuel pit heat exchanger will handle a heat load of 5.3×10^6 BTU/hr with a service water temperature of 80°F. To insure the spent fuel pit temperature will not be exceeded, a limit is placed on the system heat load.

During full core discharge the spent fuel pit temperature is limited to 150°F[3]. Under these conditions the system will handle 9.3×10^6 BTU/hr and the system heat load is correspondingly changed.

The decay heat will be calculated using Reference (4) plus 20%. The fuel assemblies will be assumed to have been irradiated at rated core power for the average burnup of the discharged fuel and decay time will be the average for the discharged fuel.

Add these References to the end of Section 3.8.

(3) FSAR - Section 9.3.1

(4) ANS - 5.1 (N18.6), October 1973

Attachment B

Spent Fuel Storage Rack Replacement

[illegible]

Spent Fuel Storage Rack Replacement

TABLE OF CONTENTS

I. INTRODUCTION

The spent fuel pool at the Ginna Plant is described in Section 9.3 of the Final Safety Analysis Report (FSAR) and has a capacity for storing 210 fuel assemblies. There are currently 56 fuel assemblies in the pool, and it is anticipated that 36 more assemblies will be placed in the pool at the Spring 1976 refueling outage.

RG&E presently has a contract with Nuclear Fuel Services, Inc. for fuel reprocessing. RG&E shipped the initial core loading of fuel, 121 fuel assemblies, to NFS in the Spring of 1973. However, in response to the present lack of storage space at NFS and to insure the capability for full core discharge following the refueling in March 1977, RG&E is planning to replace existing racks, in which fuel assemblies are stored with 21 inch center-to-center spacing, with new racks with a mean distance between centers of fuel of 12-1/2 inches. The number of spent fuel storage positions will be increased to 595. This will allow RG&E to store all spent fuel assemblies from Ginna through 1985 and have the capability to unload all fuel from the reactor vessel.

The replacement of the spent fuel racks is planned for the Fall of 1976. The rack replacement will be performed while there is water and spent fuel in the pool. After approximately half of the existing racks are replaced, the spent fuel in the pool will be transferred to the new racks, and replacement of

the remaining racks will be completed. Applicable safety and design criteria, including seismic capability for all racks containing fuel, will be satisfied at all steps in the rack-replacement procedure. The use of divers is anticipated to facilitate removal and installation operations.

Under the present fuel management plan, decay heat removal requirements will be less than the design capability of the Spent Fuel Pool Cooling System as described in the FSAR until after the refueling scheduled for 1981. The Spent Fuel Pool Cooling System is presently capable of removing 5.3×10^6 Btu/hr under Normal Refueling Conditions and 9.3×10^6 Btu/hr under Full Core Discharge Conditions. Under the proposed change to the Technical Specification, the decay heat load from the fuel stored in the Spent Fuel Pool will be limited to these values until modifications can be made to increase the Spent Fuel Pool heat removal capability.

All design, analysis, and fabrication are being performed under direction of Wachter Associates, Inc. The nuclear analysis is being performed for Wachter Associates, Inc. by Pickard, Lowe, and Garrick, Inc. Installation will be performed under the technical guidance of Wachter Associates, Inc.

II. DESCRIPTION OF NEW DESIGN

A. Spent Fuel Rack

The present spent fuel racks will be replaced by new spent fuel racks that will increase the storage capacity to 595 assemblies.

The new spent fuel rack is a modular design arranged in a checkerboard pattern. The inherent strength of this rack design is its honeycomb box structure arrangement. (Every box in the module is solidly fastened to adjacent boxes, thus resulting in an extremely rigid structure.)

The rack assemblies are made up of a repeating array of square stainless steel boxes. Alternate boxes in the checkerboard pattern are designed to contain spent fuel assemblies. The remaining boxes will contain pool water. The stainless steel boxes are approximately 13-1/2 feet long, 8.25 inches square (on the inside) and 0.090 inch thick. The lower end of each box contains a horizontal plate, with a circular hole in the center, to hold the spent fuel assembly.

There are three types of rack modules; Type A which contains 70 fuel assembly locations, Type B which contains 56 fuel assembly locations, and Type C which contains 49 fuel assembly locations. These units are structurally

identical and differ only in the number of boxes used to construct them. The empty weight of the Type A unit is approximately 20,700 pounds, the empty weight of the Type B unit is approximately 16,600 pounds, and the empty weight of the Type C unit is approximately 14,700 pounds.

B. Rack Arrangement

The arrangement of the racks in the spent fuel pool consists of seven Type A racks and one each of Type B and C, as shown in Figure 1. This arrangement provides space at the south end of the pool for the shipping cask and other items needed in fuel handling operations. The racks will occupy less space than is presently occupied by spent fuel racks.

C. Rack Base

A stainless steel I-beam base will be installed in the pool for each rack module. These bases are provided with leveling pads which are interconnected mechanically to each other and are laterally supported off the wall by means of large bearing pads. (See Figure 2.) The racks are solidly bolted to the rack bases. (See Figure 3.)

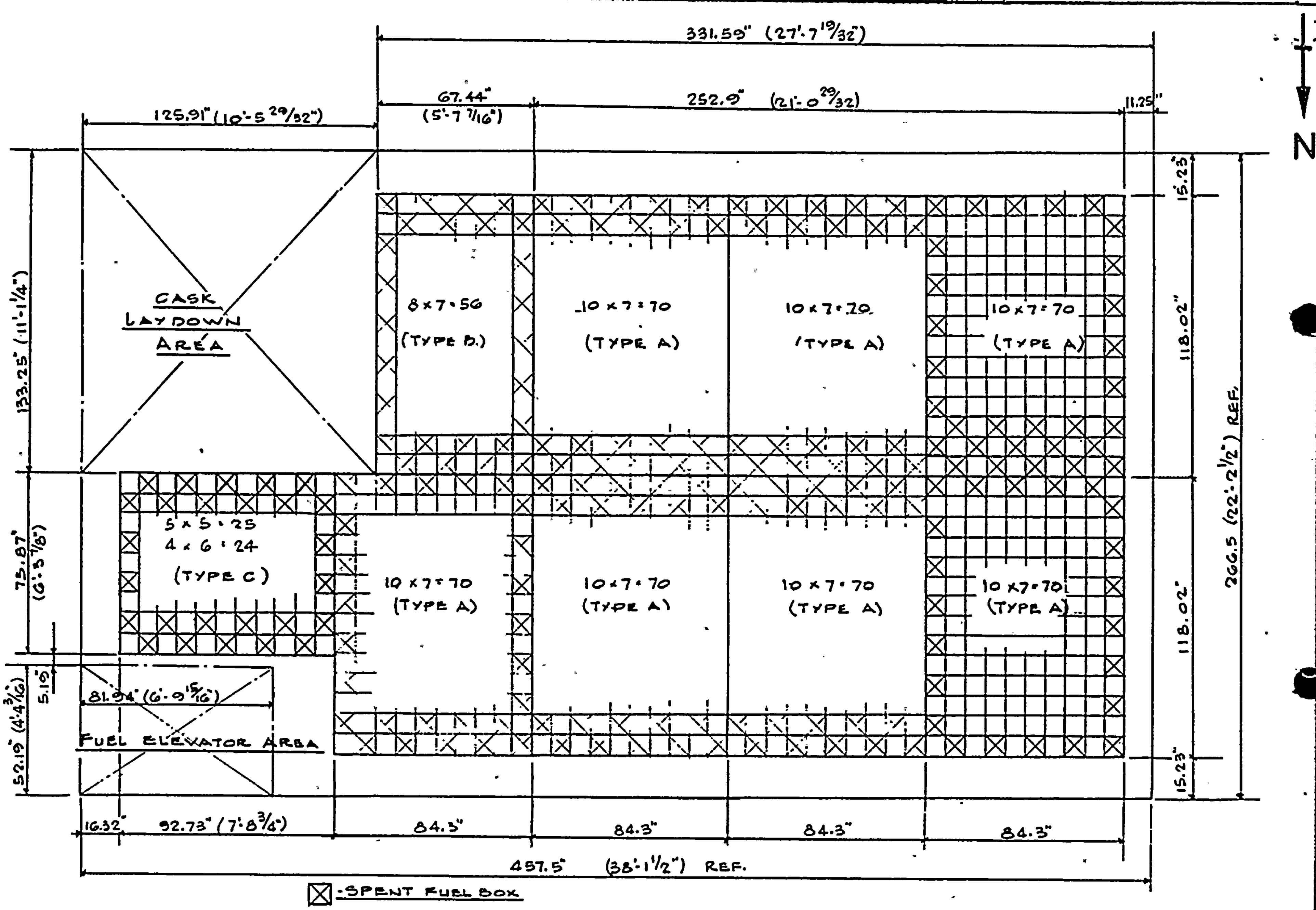

FIGURE 1

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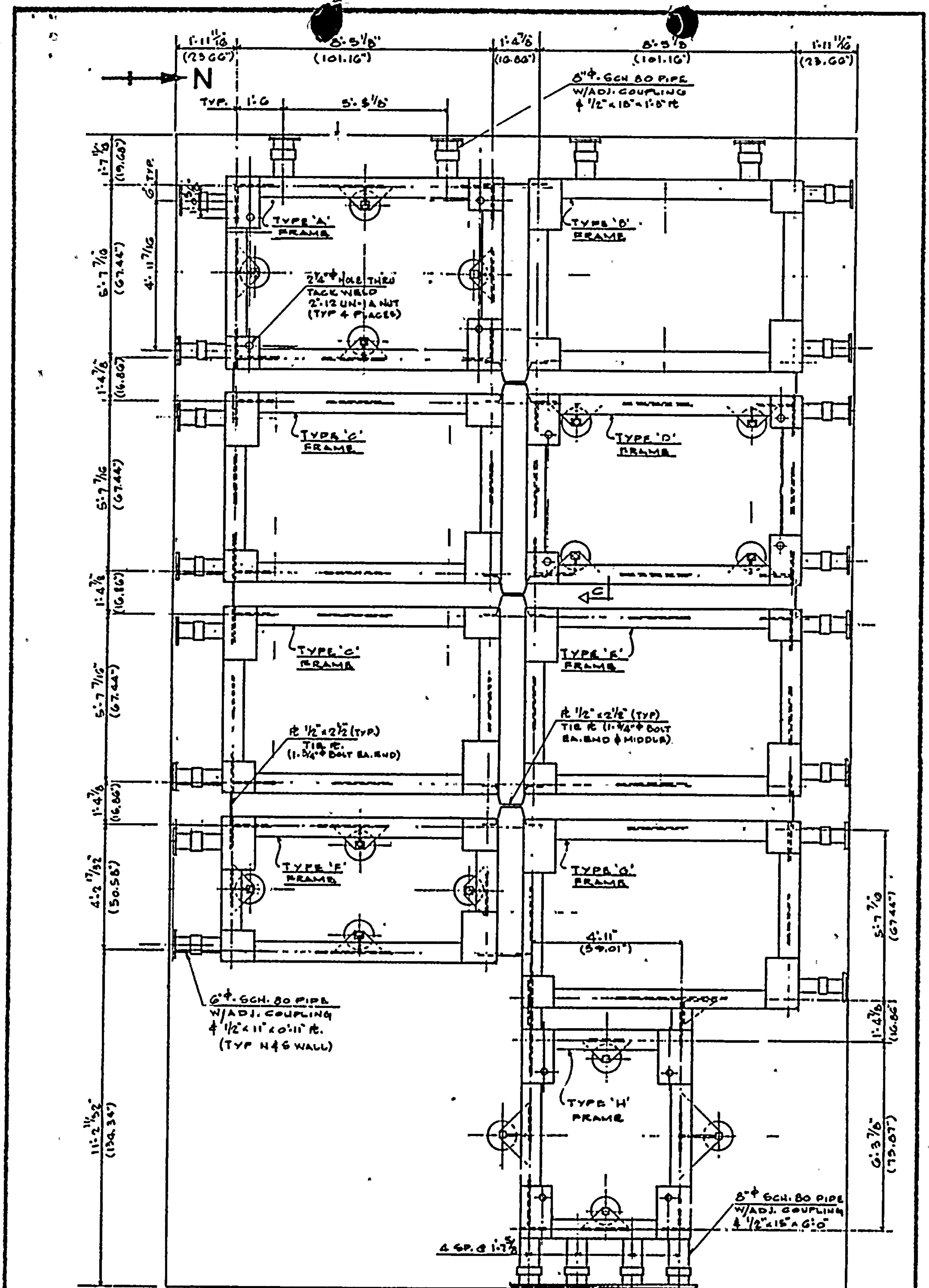


FIGURE 2

WACHTER ASSOCIATES INC.

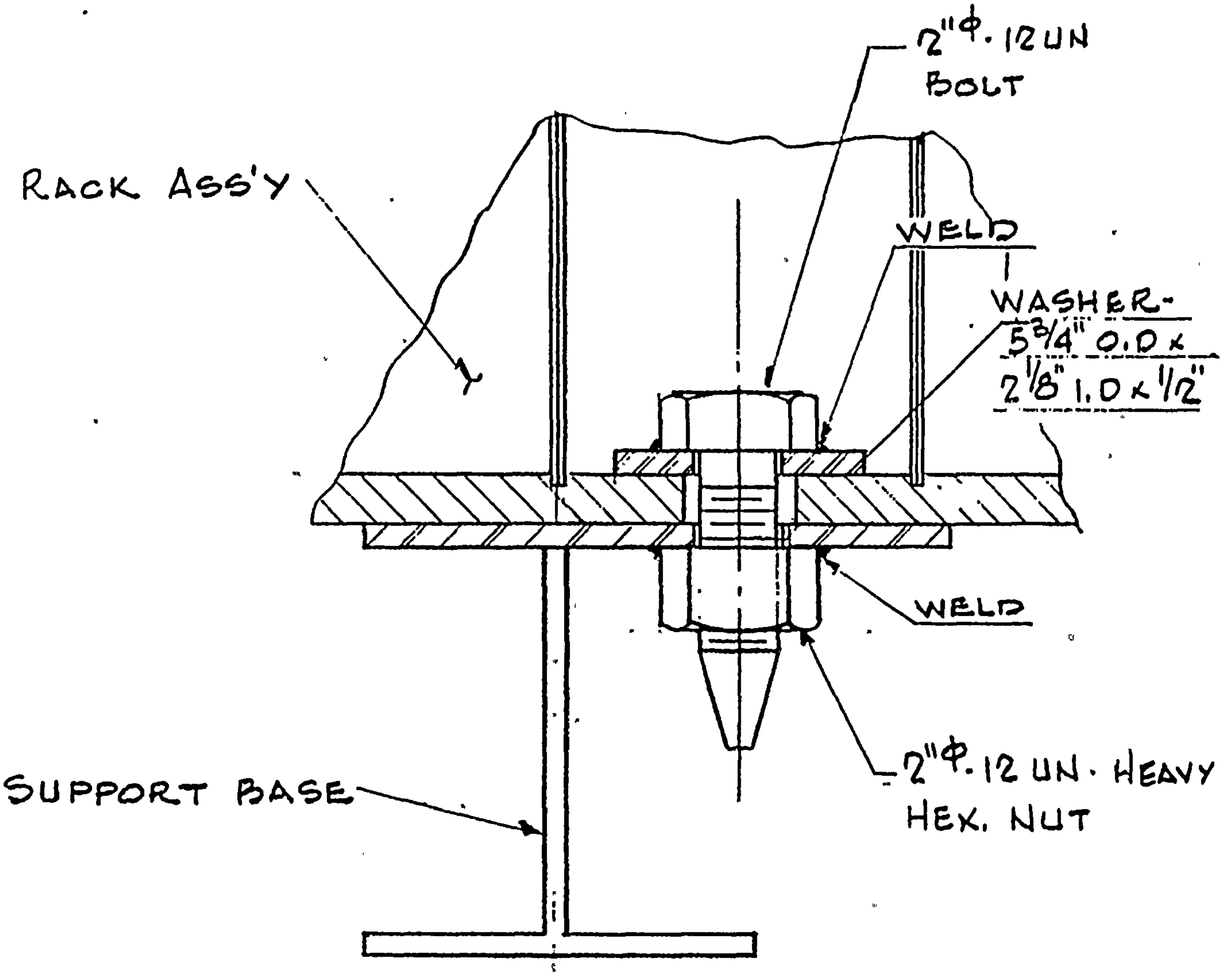


FIGURE 3

WACHTER ASSOCIATES INC.

12-12-75

III. EVALUATION OF NEW DESIGN

Criteria for the design and performance of spent fuel storage systems are defined by ANSI Standard N 18.2 - 1973 and USNRC Regulatory Guide 1.13. The new spent fuel rack satisfies these criteria, as described below. Where compliance with a criterion is not affected by the modification, the criterion is not listed.

A. ANSI N 18.2 - 1973

5.7.4 Performance Criteria

5.7.4.1 "The design of spent fuel storage racks and transfer equipment shall be such that the effective multiplication factor will not exceed 0.95 with new fuel of the highest anticipated enrichment in place assuming flooding with pure water ... Credit may be taken for the inherent neutron absorbing effect of materials of construction or, if the requirements of 5.7.5.10 are met, for added nuclear poisons."

Assuming new fuel with an enrichment of 3.5 w/o, the effective multiplication factor of the new rack design is less than 0.8871, including uncertainties. Ginna Technical Specification 5.3.1.c limits fuel enrichment to no more than 3.5 w/o of U-235. Credit is taken in the calculations for the inherent neutron absorbing effect of the stainless steel used in the structure and for boxes that are utilized as neutron poison. The method of calculation is presented in Section V. The requirements of Criterion 5.7.5.10 are satisfied as discussed subsequently.

5.7.4.2 "Fuel handling system facilities shall be designed to prevent damage to fuel assemblies while in storage or during transport from one location to another."

Each fuel assembly is stored in a stainless steel box, which physically separates that fuel assembly from all other fuel assemblies. The stainless steel box will be strong enough to prevent damage to the contained fuel assembly in the unlikely event that another fuel assembly should be dropped anywhere on top of the spent fuel racks.

The new rack design contains no protuberances that could cause damage to a fuel assembly being lowered into or being lifted out of a storage position. Adequate lead-ins are provided at the top of the boxes.

5.7.4.3 "The fuel storage pool capacity shall accommodate at least one shipping cask and one complete core in addition to the maximum number of fuel assemblies normally stored in the pool. Consideration should be given to potential for highly radioactive components which may require storage in the pool."

The new rack design will provide the capability to store all spent fuel assemblies from approximately 11 years of reactor operation and still retain the capability to accommodate one shipping cask and removal of the complete core from the reactor vessel. During refueling periods, and whenever the shipping cask is not in the pool, the cask area will be available to store radioactive components or to perform underwater inspection of or underwater mechanical operations on radioactive components. There is also space between the spent fuel racks and the walls of the pool which is available for longer-term storage of radioactive components.

5.7.4.6 "Suitable provisions shall be made in the design of the fuel storage pool cooling system to permit installation of instrumentation to monitor system performance."

The pressure and flow of service water through the Spent Fuel Pool heat exchanger and the temperature and pressure of Spent Fuel Pool water circulating through the Spent Fuel Pool heat exchanger are measured and indicated locally. The Spent Fuel Pool water temperature is measured and a high temperature alarm is actuated in the control room if the Spent Fuel Pool water temperature exceeds 115°F. The Spent Fuel Pool water level is also measured and a High/Low alarm is actuated in the control room if the water level exceeds preset values.

5.7.5 Mechanical Design Criteria

5.7.5.1 "The fuel storage pool and storage racks shall be designed to accommodate, within applicable code stress limits, normally imposed loads due to half the Design Basis Earthquake."

This criterion is satisfied as described in Section VII of this report.

5.7.5.2 "The fuel storage pool and storage racks shall be designed so that normally imposed loads plus loads imposed by the Design Basis Earthquake will not cause failure. Plastic deformation may take place but with a substantial margin to that which might result in failure."

The fuel storage pool is founded on sound rock. The new spent fuel storage racks are capable of withstanding loads imposed by the Design Basis Earthquake without plastic deformation of the racks and without damage to spent fuel assemblies. The bearing loads are sufficiently low to prevent damage to the stainless steel liner of the spent fuel pool and supporting concrete. The reinforced concrete structure of the pool is capable of transmitting these loads to the rock without plastic deformation of the pool structure.

5.7.5.3 "Lifting and transport equipment of the fuel handling system shall be designed to prevent dropping of fuel assemblies. Heavy loads shall not be carried over stored fuel assemblies. The design shall prevent lifting a fuel shipping cask over fuel storage racks."

These provisions are included in the initial design. The outer envelope of the new spent fuel racks is entirely within the outer envelope of the existing spent fuel racks and, therefore, compliance with these criteria is not affected.

5.7.5.4 "Fuel storage racks shall physically prevent placing more than one fuel assembly in a single storage location; specified minimum center-to-center distances between individual fuel assemblies shall be maintained to meet requirements of Section 5.7.4.1."

The new rack design permits only one fuel assembly to be inserted into a storage box. Minimum center-to-center spacings between fuel assemblies are maintained by the rack structure.

5.7.5.5 "Fuel storage rack design shall prevent geometric changes due to environmental conditions characteristic of this site. The design shall be stable against tipping with provisions to prevent unplanned movement of the fuel or the racks."

The geometry of the new rack design cannot be changed by seismic events, nor by other environmental conditions characteristic of the site. The rack design is stable against tipping. Each stored fuel assembly is completely surrounded by a relatively close-fitting box.

5.7.5.8 "The fuel storage pool and refueling canal shall have provisions, such as a watertight liner, to prevent leakage of pool water."

A stainless steel liner is provided. The new spent fuel racks and their seismic supports are designed to limit local mechanical loadings on the pool liner to prevent damage to the liner. In addition, installation of the new racks does not require complete removal of the existing welded rack supports and the racks will not be welded to the existing liner, thus precluding possible damage to the liner during installation. The new racks are supported as described in Section II-C of this report.

5.7.5.9 "The fuel storage pool minimum depth shall be determined by dose considerations at the top of the pool considering irradiated fuel or components stored in the pool or in transit and radioactive contaminants in the pool water."

The depth of water over the spent fuel is unchanged by the new rack design. A radiological evaluation of the new rack design is presented in Section VIII of this report.

5.7.5.10 "Fuel storage racks using nuclear poisons additional to those inherent in the structural materials shall be designed and fabricated in a manner to prevent inadvertent removal of the additional poisons by mechanical or chemical action. Prior to installation of the additional nuclear poisons, the quantity and effectiveness of the additional poisons shall be verified. Effectiveness of the additional poisons may be checked by isotopic analysis. Provisions shall be made to permit periodic inspection or verification or both, thereafter."

The proposed spent fuel rack design does not employ nuclear poisons in addition to those inherent in the structural materials.

5.7.5.13 "Provisions shall be made to accommodate the necessary heavy equipment loads in the fuel storage pool without subjecting the pool liner to mechanical damage."

The bearing loads on the pool liner are low and will not cause mechanical damage to the liner. Bearing loads are described in Section VII of this report.

B. <u>NRC REGULATORY GUIDE 1.13</u>

1. "The spent fuel storage facility (including its structures and equipment except as noted in Section 6 below) should be designed to Category I seismic requirements."

The spent fuel pool is designed to Category I seismic requirements as described in Section 5.1.2.4 of the FSAR. The new spent fuel racks are also designed to Category I seismic requirements as described in Section VII of this report and as discussed in the responses to Criteria 5.7.5.1 and 5.7.5.2 of ANSI 18.2 - 1973.

5. "The spent fuel storage facility should have the following provisions with respect to the handling of heavy loads, including the refueling cask:

   a. Cranes capable of carrying heavy loads should be prevented, preferably by design rather than by interlocks, from moving into the vicinity of the pool, or

   b. The fuel pool should be designed to withstand, without leakage which could uncover the fuel, the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted. If this latter approach is followed, design provisions should be made to prevent this crane, when carrying heavy loads, from moving in the vicinity of the stored fuel."

Section IX-B of this report documents that these aspects of the facility are not affected by replacement of the spent fuel racks because the outer envelope of the new spent fuel racks is within the outer envelope of the existing racks.

IV. INSTALLATION

The replacement of racks will be accomplished prior to the Spring of 1977 refueling outage. The rack replacement will be performed while water is in the pool. After approximately half the existing racks are replaced, the 92 spent fuel assemblies in the pool will be removed into the new racks, and the replacement of the remaining racks will be completed. Applicable safety and design criteria will be satisfied in all steps of the rack replacement procedure. It is planned to use divers to assist in the above operation.

V. NUCLEAR ANALYSIS

A. Methods of Analysis

The LEOPARD[1] computer program was used to generate macroscopic cross sections for input to four energy group diffusion theory calculations which are performed with the PDQ-7[2] program. LEOPARD calculates the neutron energy spectrum over the entire energy range from thermal up 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 analytic 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].

In addition, 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 these 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

affected the accuracy of the PLG-modified LEOPARD program for calculations of slightly enriched UO_2 systems.

The UO_2 critical experiments chosen for benchmarking include 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 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 LEOPARD results is shown in Table V-1 for the 27 measured criticals chosen as being directly applicable for benchmarking the model for spent fuel pool calculations. The average calculated k_{eff} is 0.9979 and the standard deviation from this average value is 0.0080 Δ k. Reference 5 raised questions concerning the accuracy of the measured bucklings reported for the experiments number 12 through 19. If these data 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 Δ k.

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 calculating the subcritical margin of the proposed spent fuel pool arrangement.

As a specific demonstration of the accuracy of the calculational model used for the spent fuel pool calculations, the combined LEOPARD/PDQ-7 model has been used to calculate seven measured 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 spent fuel pool design. Experiments including soluble boron are included in this demonstration since we are primarily interested in 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.

These LEOPARD/PDQ-7 calculations, shown in Table V-2, result in a calculated average k_{eff} of 0.9922 with a standard deviation about this value of 0.0014 Δ k. These results together with the previously discussed LEOPARD results demonstrate that the proposed LEOPARD/PDQ-7 calculational model can calculate the reactivity of the proposed spent fuel pool arrangement with an accuracy of better than ± 0.01 Δ k.

B. Evaluation of Reference Design

The PDQ-7 program is used in the final predictions of the reactivity of the spent fuel storage pool. The calculations are performed in four energy groups and take into account all the significant geometric details of the fuel bundles, 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 two identical stainless steel boxes. The specific geometry and dimensions of this basic cell are shown in Figure V-1.

The calculational approach is to use the basic cell to calculate the reactivity of an infinite array of uniform spent fuel racks and to account for any deviations of the actual spent fuel rack array from this assumed infinite array as perturbations on the calculated reactivity of the basic cell. The effects of mechanical tolerances are also treated as perturbations on the calculated reactivity of the basic cell. The fuel bundles were assumed to be unirradiated with a U-235 enrichment of 3.5 w/o which is higher than any anticipated reload enrichment for the Ginna core. Most of the calculations were performed at a uniform pool temperature of 80°F, but the reactivity effects of pool temperature are also taken into account as a perturbation on the basic cell calculations.

The reference basic cell calculation is performed with the minimum dimension on all the stainless steel boxes which results in a $k_\infty = 0.8779$. Other tolerances on the geometric array representing the racks are treated as perturbations on this reference basic cell calculation.

The stainless steel fuel and water boxes are nominally .090 inches thick with a tolerance of ± .004 inches. Assuming a worst case in which all boxes were at the minimum thickness of .086 inches the k_∞ of the basic cell is .8806. Therefore, the maximum perturbation on the reactivity of the basic cell due to variations in the stainless steel box thickness is +.0027 Δ k.

With the fuel bundles located in their most reactive positions inside the stainless steel boxes, the k_∞ of the basic cell is .8807. Thus, the perturbation on the basic cell reactivity due to positioning uncertainties is + .0028 Δ k.

Most of the calculations with the basic cell geometry utilized a 50 x 25 two-dimensional array of mesh points. To test the adequacy of this mesh description a calculation was run with a 100 x 50 mesh size and the resulting k_∞ was .8777. Thus the perturbation on the basic cell due to mesh spacing effects is - .0002 Δ k.

The k_∞ of the basic cell as a function of temperature is shown in Figure V-2. With a maximum pool temperature of 200°F under the worst possible conditions the k_∞ is 0.8838, which results in a perturbation due to temperature effects of +0.0059 Δ k. Although the overall steady state reactivity temperature coefficient of the spent fuel pool is positive, the temperature coefficient of the fuel bundles is negative.

The basic cell was also used to evaluate the reactivity effect of axial neutron leakage. Using an axial buckling based on a 142 inch active fuel length with a total reflector savings of 15 cm, the calculated k_∞ of the basic cell is .8759. Thus the reactivity pertubation due to axial neutron leakage is - .0020 Δ k.

A summary of the perturbations to the basic cell reactivity calculation is shown in Table V-3. Thus the calculated reactivity of the spent fuel pool with 595 unirradiated bundles with 3.5 w/o U-235 is .8871 for a pool temperature of 200°F.

C. Uncertainty Considerations

In Section V.A it was demonstrated that the uncertainty in the calculated k_{eff} with the model utilized for criticality calculations is less than ± 0.01 Δ k. It will now be demonstrated that there are a number of conservatisms in the model's representation of the spent fuel pool such that these conservatisms more than compensate for the uncertainty in the calculational model. Therefore, the effective multiplication factors presented in Section V.B are conservative even when the effects of model uncertainties are included.

The basic cell calculations of k_∞ apply to an infinite array of racks containing unirradiated fuel bundles with no burnable poisons and no net radial neutron leakage. The maximum reload batch size anticipated for the Ginna core is less than or equal to 40 bundles. Therefore even if the entire core were to be discharged shortly after the start of a fuel cycle, there would be at most 40 unirradiated fuel bundles in the spent fuel pool. In such a situation there would be significant neutron leakage from the 40 unirradiated bundles to surrounding irradiated bundles or to empty fuel locations or to the water reflector. It is conservatively calculated that the resulting radial neutron leakage would reduce the calculated reactivity of the basic cell by .0102 Δ k.

The spacer grids utilized in the design of the Ginna fuel bundles contain inconel spacers which result in parasitic neutron absorption which is not included in the basic cell calculations. The spacer grids are calculated to reduce the k_∞ of the basic cell by .0086 Δ k.

The inherent conservatisms in the analytical model are such as to reduce the calculated k_∞ of the basic cell by at least .0188 Δ k. The reduction in k_∞ is nearly twice the possible increase in the k_∞ of the basic cell due to uncertainties in the analytical model. Therefore the multiplication factor of the spent fuel pool is < .8871, the value reported in Section B above, and Criterion 5.7.4.1 in ANSI N 18.2-1973 is satisfied.

D. Additional Considerations

These analyses take credit only for the inherent neutron absorbing properties of the type 304, stainless steel boxes which are the principal structural components of the spent fuel racks. Fe, Ni, Cr, and Mn account for 99% of the composition of type 304 stainless steel and these are the only constituents which are considered to absorb neutron in these analyses. Other constituents, including impurities, will result in some small additional neutron absorption which will slightly increase the subcriticality of the rack.

The construction of the spent fuel racks is such that a dropped fuel bundle cannot under any conceivable circumstance penetrate and occupy a position other than a normal fuel storage location. Therefore a dropped fuel bundle will end up in a final position that is somewhere between vertical and horizontal on top of the racks. The only positive effect of such a bundle on the reactivity of the rack would be by virtue of a reduction in axial neutron leakage from the rack. Since the calculations reported here show the total axial neutron leakage effect to be .0020 Δ k, a dropped fuel bundle would not have any significant effect on the reported maximum possible reactivity of the spent fuel storage rack.

TABLE V-1

Case+ Number	Reference Number	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	3.86	10.18	0.7620	0.8594	0.04085	1.1938	63.26	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	10.38	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	0.9987
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	4.069	2.55	9.46	1.1278	1.2090	0.0406	1.5113	88.0	0.9859
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	0.9872
18	16	3.037	2.64	9.28	1.1268	1.2701	0.07163	1.555	50.75	0.9946
19	16	3.037	8.16	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	69.1	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

\+ Cases 1 through 19 are with stainless stell clad, Cases 20 through 27 are zircalloy clad.

TABLE V-2

WESTINGHOUSE UO_2 CRITICAL EXPERIMENTS

(References 6 and 7)

Expt	Boron (ppm)	H_2O/UO_2 (Volume)	Pitch (In)	k_{eff} (PDQ-7)
1	0	1.49	.600	.9905
2	0	2.42	.690	.9949
3	0	4.35	.848	.9921
4	0	6.21	.976	.9918
5	306.0	1.49	.600	.9912
6	536.4	1.49	.600	.9925
7	727.7	1.49	.600	.9926

TABLE V-3

Reactivity Perturbations on the Reference Basic Cell Calculation

Description of Reactivity Perturbation	Reactivity Effect, Δ k
Mechanical tolerance spacing on stainless steel boxes	0.00
Fuel position within stainless steel boxes	+ .0028
Mechanical tolerances on stainless steel box walls	+ .0027
Mesh effects	- .0002
Temperature increase to 200°F	+ .0059
Axial neutron leakage	- .0020
Total perturbation on basic cell reactivity calculation	+ .0092

FIGURE V-1

NOTE: Boundary Condition at the Top of this Figure is 180° Rotational Symmetry

RGE RACK REFERENCE DESIGN

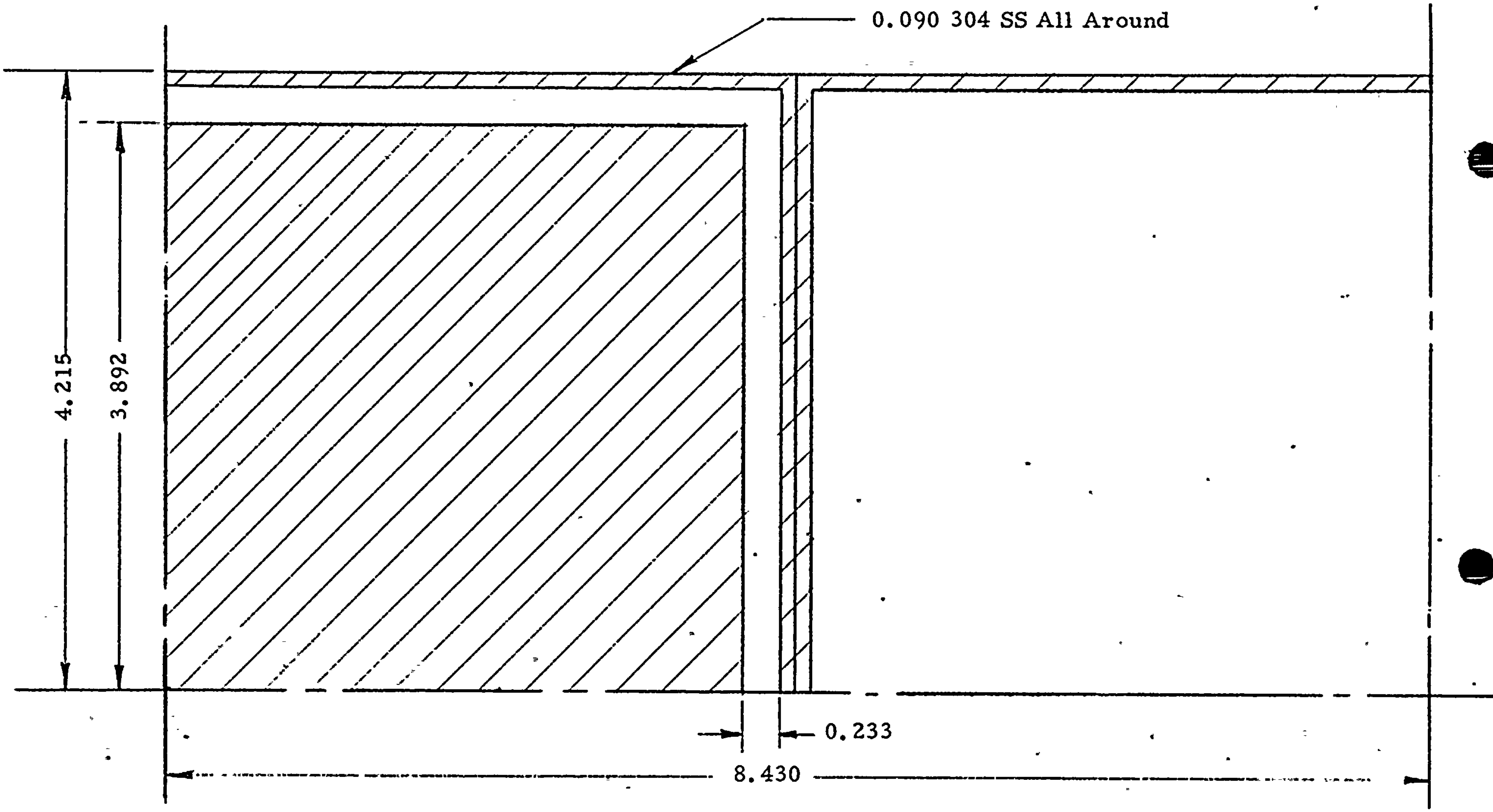
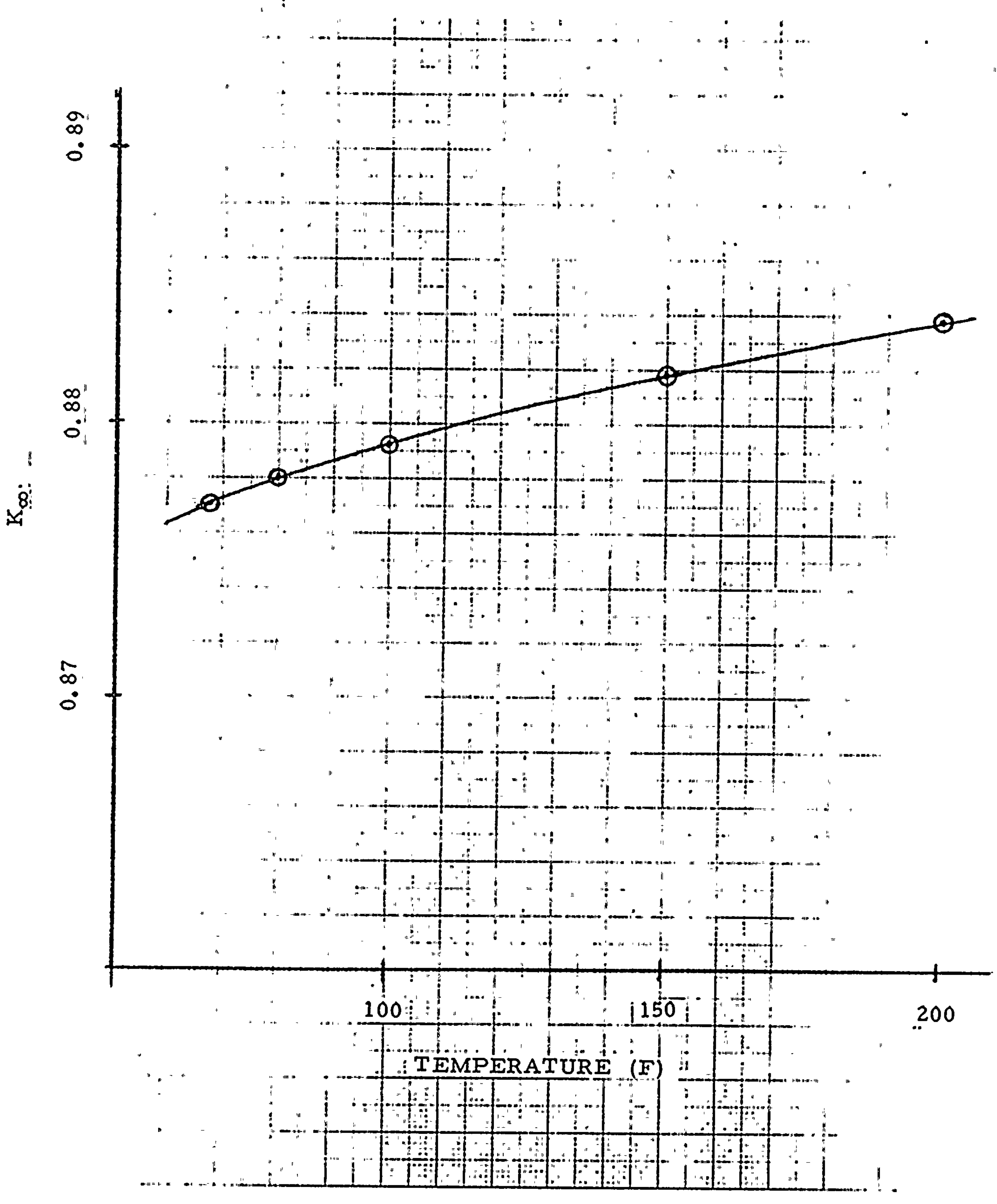


FIGURE V-2
EFFECTIVE MULTIPLICATION
VS.
TEMPERATURE



VI. THERMAL - HYDRAULIC ANALYSIS

A. Heat Removal Requirements

The heat removal criteria of the Spent Fuel Pool Cooling System (SFPCS) are given in Section 9.3.1 of the Ginna FSAR and are that the system must be capable of maintaining the Spent Fuel Pool (SFP) temperature less than or equal to 120°F during Normal Refueling operations and less than or equal to 150°F during Full Core Discharge situations.

Normal Refueling operations are conducted annually with nominally 40 fuel assemblies (one-third of the core) being removed from the core and placed in the SFP.

Full Core Discharge occurs when all the fuel in the reactor (121 fuel assemblies) is placed in the SFP. The full core will be discharged once every ten years to enable inspection of the lower reactor internals. Full core discharge may also occur on other occasions when it is deemed necessary to remove the core to perform maintenance on the reactor lower internals or pressure vessel.

B. Service Water Temperature

The Spent Fuel Pool (SFP) heat exchanger transfers heat from the SFP water to the service water. The Service Water System is discussed in Section 9.6.2 of the Ginna FSAR.

The temperature of the service water going into the SFP heat exchanger is a controlling factor in determining the heat transfer capability of the SFP cooling system. The service water temperature is the same as the intake (lake) water temperature except during the winter months when recirculation is used as necessary to maintain a water temperature of approximately 37°F.

Table VI-1 illustrates the monthly average of the daily minimum, average, and maximum intake water temperatures.

Table VI-2 presents lists of the minimum and maximum intake water temperatures that occur at any time during each month.

The intake water temperature has been recorded since December 1969. The data show the following:

1. the instantaneous daily maximum temperature has exceeded 80°F three times and then only by a maximum of two degrees.

2. the monthly average of the daily maximum temperatures has not exceeded 75°F.

3. the monthly average of the daily average temperatures has not exceeded 73°F.

The service water temperature to the inlet of the SFPCS heat exchanger can therefore be assumed to be 80°F or less.

C. Analysis of Heat Removal System

The SFPCS consists of a single loop containing a pump and heat exchanger. Water is drawn from the SFP by the SFP pump, forced through the heat exchange, and returned to the SFP. The heat exchanger is cooled by the Service Water. Approximately 10% of the water from the SFP bypasses the heat exchanger and is passed through a demineralizer and filter.

The design capabilities of the SFPCS were calculated for 120°F (maximum Normal Refueling temperature) and 150°F (maximum Full Core Discharge temperature). A service water flow of 700 gpm at 80°F was assumed with a SFP outlet flow of 610 gpm and with only 550 gpm flowing through the SFPCS heat exchanger. Under these conditions, the heat exchanger, with design fouling, will transfer 5.3×10^6 BTU/hr with a SFP outlet temperature of 120°F and 9.3×10^6 BTU/hr with a SFP outlet temperature of 150°F.

The impact of the proposed modification on the heat load has been evaluated for Normal Refueling operation and the Full Core Discharge. In both cases the decay heat was calculated from the ANS - 5.1, N18.6 standard plus 20% assuming finite irradiation. Table VI-3 illustrates the results of these calculations using the following assumptions:

a. all fuel assemblies irradiated at rated core power for the entire design burnup of the fuel assemblies except for those assemblies presently in the SFP. These assemblies were assumed to be irradiated a rated core power for the actual average assembly burnup.

b. refueling takes place annually.

c. one-third of the core (40 assemblies) is discharged annually except for the 1976 refueling when 36 assemblies will be discharged.

Full Core Discharge -

a. the emergency outage occurs one year after the last refueling and consists of a full core unload (121 assemblies) into the SFP in addition to the assemblies already in the pool from previous refuelings.

b. a full core unload consists of 3 regions with burnups of one-third, two-thirds, and design assembly burnup.

As can be seen from Table VI-3 the heat load on the SFPCS decreases as the time between reactor shutdown and placement of the fuel in the SFP is increased. The SFPCS is capable of maintaining SFP temperature below 120° and 150°F for several years without requiring unreasonable decay time.

During this period, modifications to the SFPCS will be considered that will increase the cooling capability of the system. Until these modifications are complete the SFP heat load will be limited to 5.3×10^6 BTU/hr and 9.3×10^6 BTU/hr respectively.

The calculations summarized above are conservative for the following reasons:

a. No credit is taken for heat loss by evaporation from the pool surface. Heat is assumed to be removed only by the SFPCS heat exchanger.

b. No credit is taken for the heat capacity of the SFP water. Transient calculations accounting for this effect would allow greater instantaneous heat loads without exceeding required temperature limits.

c. Measurements have shown that it is possible to have greater than design service water flow through the SFPCS heat exchanger. The calculations assume design flow; greater flow would result in larger heat transfer.

d. Refuelings are scheduled for March or April of each year when the lake water temperature is less than 45°F. Under these conditions, the SFPCS heat exchanger can be operated at higher heat fluxes without exceeding a SFP temperature of 120°F. Similar conditions would exist for a scheduled unloading of the core which took place during a normal refueling.

D. Cooling Analysis of Individual Fuel Assemblies

At present, water is returned to the SFP from the SFPCS heat exchanger through a discharge pipe entering the pool near the center of the south wall. Water enters the SFPCS through another pipe also located on the south wall. To insure proper cooling of the fuel assemblies in the proposed spent fuel rack modification, the discharge pipe will be rerouted to run along the west wall of the SFP where it will discharge water in the wall-rack space just above the seismic supports. All of the fuel rack base I-beams have holes cut in them to provide 50% free-flow area; this amounts to 7.38 ft^2 alone in the I-beams facing the west wall. Another 1.64 ft^2 is provided by the 2" beam-to-floor gap (2" minimum, specified as 3"), although its presence is not crucial.

The area inside the I-beams under the rack boxes is completely open and free of obstruction except for the relatively minor effects of the jack screws and their supports. The bottoms of the fuel boxes and poison boxes are flush with others so that the only pressure losses inside the I-beams are flow-branching losses which are minor compared to passing through (and under) the I-beams.

Each fuel assembly's flow will depend upon its heat dissipation rate and the total pressure loss experienced by the base flow reaching its inlet (lower nozzle) location, which in turn depends slightly upon other fuel assembly heat rates and flows. Fuel assemblies having higher (than average) heat dissipations draw higher flow rates, but not enough to prevent a higher outlet temperature. The flows from all the fuel assemblies mix above the fuel racks and move toward the south wall outlet.

The cooling of the individual assemblies has been analyzed assuming the worst case conditions. A full core discharge situation was assumed with a pool heat load of 9.3×10^6 BTU/hr, a pool outlet temperature of 150°F, and a pool flow of 610 gpm. The hottest fuel assemblies were assumed to be located near the east wall since the water reaching these fuel assemblies experiences the greatest pressure loss by having to pass through either 7 or 9 I-beams. The fuel

assemblies having average heat dissipation were assumed to occupy racks in the center of the pool, where their cooling water passes through 3 I-beams.

The cooling analysis accounts for the pressure losses due to the I-beams and the 2-inch beam/floor gap. Flow branching Δ P's are negligible. Fuel assembly pressure losses are also accounted for and were found to be the dominant factor in comparison to the I-beam losses which were calculated on the basis of decreasing flow and velocity head due to branching.

By subtracting the pool-average fuel assembly Δ P and its associated I-beam losses, the hottest fuel assembly Δ P and its associated I-beam losses are calculated as a function of its fuel assembly flow; it increases when plotted against the ratio, hottest-fuel assembly-flow/pool-average-fuel assembly-flow. The driving differential for hottest-fuel assembly flow in comparison to pool-average-fuel assembly flow is the difference in the average water densities in the two fuel assemblies times the active fuel height, and decreases with increasing flow ratio. The intersection of these two curves defines the operating point of the hottest fuel assembly. Results show the hottest fuel assembly with an outlet temperature of less than 155°F.

The maximum cladding temperature, accounting for the film- ΔT, is less than 160°F. Considering that the local saturation temperature is 242°F, the calculated temperatures for these worse case conditions are acceptable.

TABLE VI - 1

MONTHLY AVERAGE INTAKE WATER TEMPERATURE (°F)
VERSUS TIME OF YEAR
GINNA STATION

	1970			1971			1972			1973			1974			1975		
	MIN	AVG	MAX	MIN	AVG	MAX	MIN	AVG	MAX	MIN	AVG	MAX	MIN	AVG	MAX	MIN	AVG	MAX
JAN	34.3	35.0	35.5	32.4	32.9	34.3	32.8	33.6	35.1	33.3	33.7	34.6	35.0	35.8	37.9	36.7	37.7	39.2
FEB	31.4	31.8	32.5	32.7	33.2	34.1	32.0	32.1	32.8	33.1	33.3	33.8	32.6	33.0	33.3	36.2	36.9	37.9
MAR	32.3	32.8	33.2	NR	NR	NR	32.7	32.9	33.8	35.2	35.8	36.7	35.0	35.1	36.2	36.5	36.7	37.0
APR	40.7	42.1	43.3	NR	NR	NR	35.5	36.0	37.0	38.8	39.3	40.1	40.0	41.4	43.5	39.1	39.6	40.3
MAY	41.8	42.8	43.7	42.5	43.1	44.2	NR	NR	NR	42.8	43.4	44.0	45.2	46.2	47.1	49.4	51.1	52.2
JUN	55.2	56.8	58.3	45.9	48.1	50.5	48.2	50.0	51.7	48.9	51.7	54.3	51.4	53.6	55.8	56.1	59.3	62.0
JUL	60.9	62.9	64.3	62.0	64.3	66.6	62.6	64.1	65.6	61.5	64.6	67.5	61.7	64.6	66.5	69.9	72.7	74.6
AUG	69.9	71.7	74.4	63.4	65.7	67.1	60.9	64.0	65.9	63.1	66.4	69.9	66.2	68.7	71.0	68.6	71.0	73.2
SEP	60.7	63.5	66.0	57.3	60.9	63.5	63.1	64.3	65.3	65.9	66.9	68.8	64.4	66.4	67.5	57.7	59.5	61.3
OCT	50.9	52.4	53.3	52.0	54.0	55.9	51.9	52.4	53.0	55.3	56.2	57.1	52.4	52.8	53.8	54.1	55.2	56.1
NOV	48.2	49.0	49.7	45.5	46.4	47.3	43.1	43.3	43.9	45.0	45.7	46.6	48.1	48.3	48.7	50.4	51.1	51.7
DEC	39.3	40.3	41.6	37.7	38.8	39.8	36.2	36.9	37.5	40.0	41.1	42.1	38.7	39.8	40.9	–	–	–

Intake Structure Data:
- Distance from Shore (ft): 3000
- Water Depth (ft): 30
- Average Water Withdrawal Depth (ft): 22.5

Table VI-2

MINIMUM AND MAXIMUM MONTHLY INTAKE WATER TEMPERATURE (°F)
GINNA STATION

	1970		1971		1972		1973		1974		1975	
	Min	Max	Min	Max	Min	Max	Min	Max	Min	Max	Min	Max
JAN	30	39	32	43	32	40	32	39	33	53	35	42
FEB	31	35	32	37	32	37	32	37	32	36	34	43
MAR	31	35	NR	NR	32	39	33	41	34	48	34	39
APR	39	45	NR	NR	33	42	36	45	36	56	34	47
MAY	40	46	40	48	NR	NR	40	48	42	52	43	64
JUN	45	64	38	60	42	55	41	63	43	63	43	72
JUL	55	69	41	71	47	73	42	72	43	72	43	79
AUG	58	82	41	72	46	69	40	76	43	75	43	79
SEP	41	75	40	73	45	68	49	78	45	78	43	68
OCT	41	59	40	66	48	58	44	60	47	61	43	60
NOV	43	54	42	51	38	58	42	48	43	52	47	54
DEC	35	48	33	43	32	41	37	46	37	44	-	-

Table VI-3

Total SFP Heat Load After Discharge Fuel is Placed in SFP

Discharge Year	Number of Fuel Assemblies Discharged	Total Heat Load to Spent Fuel Pool			
		Normal Refueling		Full Core Discharge	
		10 day decay	15 day decay	25 day decay	30 day decay
		$\times 10^6 \frac{BTU}{Hr}$	$\times 10^6 \frac{BTU}{Hr}$	$\times 10^6 \frac{BTU}{Hr}$	$\times 10^6 \frac{BTU}{Hr}$
1974	12	-	-	-	-
1975	44	-	-	-	-
1976	36	4.50	3.95	9.37	8.74
1977	40	5.30	4.70	9.74	9.11
1978	40	5.48	4.88	-	9.28
1979	40	5.66	5.06	-	9.46
1980	40	-	5.16	-	-
1981	40	-	5.23	-	-
1982	40	-	5.29	-	-

VII. SEISMIC ANALYSIS

The new spent fuel racks for the Ginna Plant are designed for a maximum seismic event of 0.2g horizontal acceleration applied simultaneously with normal (1.0g) gravity plus or minus 0.2g vertical acceleration. The earthquake loads on the rack and base structures are calculated on the basis of the largest, fully-loaded spent fuel rack. The direction of the horizontal seismic component is assumed to be in that worst-case direction which results in the maximum loads at any fuel rack corner joint. In addition, each fuel box (or cell) is designed to accommodate one fuel assembly with a rod control cluster assembly (RCCA) for a total design weight of 1,450 pounds.

The spent fuel racks are classified Seismic Category I in accordance with USNRC Regulatory Guide 1.29. They are designed for and will withstand the seismic loadings previously described. The honeycomb-like stainless steel structure of the rack not only provides a smooth, all-welded stainless steel box to protect the fuel assembly and preclude seismic damage, but also serves as a neutron absorber and will maintain the fuel in a non-critical (nuclear) array so long as the stainless steel box wall surrounds the fuel assembly.

The largest rack consists of 140 stainless steel boxes of which 70 are available for spent fuel storage and 70 are neutron flux traps. This 140 box rack is designated as a Type A rack. Two other rack geometries, the Type B containing 56 spent fuel

assemblies and the Type C containing 49 spent fuel assemblies, were necessary to accommodate the spent fuel pool dimensions and fuel storage restrictions. All calculations are based on the fully-loaded Type A fuel rack which serves as the worst case.

Each stainless steel box is securely fastened to its neighbors. The resulting honeycomb-like structure is quite stiff; in fact, the rack is stiffer than the support base I-beams. The I-beams serve as the load path to transfer seismic loads from the rack to the pool floor and walls, and also provide a redundant rack support structure. The stainless steel box configuration and thickness were selected on the basis of nuclear requirements as well as convenience in handling, shipping, stability and resistance to low frequency vibrations.

The loaded spent fuel rack (which includes water inertial effects and assumes that each fuel assembly contains an RCCA) and the base structure are capable of withstanding accident loads, including the Ginna OBE and DBE seismic requirements. If the rack structure were subjected to the DBE (0.2g) load, the stresses of all applicable structural components would not exceed the following AISC limitations:

a. Tension or compression over a gross section $\sigma T \leq 0.60 \ \sigma y$, $\sigma C \leq 0.60 \ \sigma y$, where σy is the 0.2% yield strength of the stainless steel.

b. Shear over gross section $\tau_s \leq 0.40\ \sigma_y$

c. Bending stresses - tensile and compressive $\sigma_b \leq 0.66\ \sigma_y$

d. Buckling stresses - compression only $\sigma_c \leq 0.60\ \sigma_{CR}$

where σ_{CR} is the lowest load critical buckling stress.

e. Tension or compression on solid round or square bars; $\sigma_{ST} \leq 0.75\ \sigma_y$, $\sigma_{SC} \leq 0.75\ \sigma_y$

also for bending stress of solid rectangular bars about weaker axis $\sigma_{RB} \leq 0.75\ \sigma_y$

Recognizing that yield stress σ_y and elastic modulus E_y are functions of temperature, both properties were extracted from tables in Section III of the 1974 ASME Boiler and Pressure Vessel Code. The temperature at which these properties have been selected is assumed to be 200°F. Since the water in the spent fuel pool is not expected to reach 200°F, the values used for σ_y and E_y are conservative.

Weld materials are generally considered to be identical to the base material since full-strength welds will be made in accordance with the AWS recommended sizes. In addition, all crucial structural welds were designed to and limited by the following stress values:

f. Groove weld - tensile stress $\sigma_y \leq 0.74 \; \sigma_{all}$

g. Groove weld - shear stress $\sigma_{WS} \leq 0.60 \; \sigma_{all}$

h. Fillet weld - shear stress $\sigma_{FS} \leq 0.49 \; \sigma_{all}$

where all three stress limits and the allowable limit value σ_{all} were extracted from tables in Section VIII of the 1974 ASME Code. The shear stress limit for fillet welds was generally less than the shear stress over gross section limitation and, therefore, was conservative.

The results are based on the following:

1. The racks are made from Type 304 stainless steel which has a minimum yield strength (0.2%) of 30,000 psi and a minimum tensile strength of 75,000 psi at room temperature. The values used in the stress and vibration analyses assume the pool temperature to be $\leq$ 200°F which result in a yield strength of 25,000 psi and an elastic modulus of 27.7 million psi.

2. The trapped water for the horizontal motion occupies all the rack space at water box locations and 0.6 of the rack space at the spent fuel location.

3. The trapped water for the vertical motion occupies 0.3 of the rack space at the spent fuel locations only. There is no vertical constraint at the water box location.

4. No benefit is taken for the horizontal friction forces between the bottom of the rack leveling pads and the pool floor.

5. No benefit is taken for the damping effect of the water.

The actual stress values calculated and the results of the seismic vibration analysis are:

1. Lowest fundamental (1st mode cantilever vibration) horizontal natural frequency of fully-loaded Type A fuel rack = 36 Hz.

2. The spectral acceleration taken from the Seismic Response Spectra, 20% g, Fig. 5.1.2-8 of the FSAR which corresponds to 36 Hz = 0.2g.

3. Lowest fundamental (1st mode simply-supported beam vibration) vertical natural frequency of fully-loaded Type A fuel rack = 277 Hz.

4. The spectral acceleration taken from the Seismic Response Spectra, 20% g, Fig. 5.1.2-8 of the FSAR which corresponds to 277 Hz = 0.2g.

5. Horizontal (10 x 7) Seismic Weight = 165,289 lbs.

6. Vertical (10 x 7) Seismic Weight = 126,442 lbs.

7. Submerged dead weight = 121,520 lbs. (neglecting buoyancy)

8. Compressive Stress on N-S Seismic Restraints, 5 in. Schedule 80 pipe

$\sigma c = 5,616$ psi for 0.2g (SSE)

$\sigma_c = 2,246$ psi for 0.08g (OBE)

$0.60\ \sigma_y = 15,000$ psi

$0.60\ \sigma_{cr} = 2.47 \times 10^6$ psi

9. Compressive Stress on E-W Seismic Restraints, 8 in. Schedule 80 pipe

$\sigma_c = 5,347$ psi for 0.2g (SSE)

$\sigma_c = 2,139$ psi for 0.08g (OBE)

$0.60\ \sigma_y = 15,000$ psi

$0.60\ \sigma_{cr} = 6.04 \times 10^6$ psi

10. Bearing Stress on N-S Walls of Pool (11" x 11" plate)

$\sigma_c = 282$ psi for 0.2g horizontally (SSE)

$\sigma_c = 113$ psi for 0.08g horizontally (OBE)

11. Bearing Stress on E-W Walls of Pool (12" x 19" plate)

$\sigma_c = 299$ psi for 0.2g horizontally (SSE)

$\sigma_c = 120$ psi for 0.08g horizontally (OBE)

12. Compressive stress in East Wall (Worst Case) of Pool due to Horizontal Seismic Loads (concrete compressive stresses)

f_c = 200 psi for 0.2g horizontally (SSE)

f_c = 81 psi for 0.08 horizontally (OBE)

Stress Limits of Concrete:

1. Gilbert Associates (4155 D-442-010)

   f'_c = 3000 psi minimum achieved after 28 days

2. 1963 ACI Building Code

   $f_c \leq 0.25\ f'_c$ for 100% area of load application

   $\leq 0.375\ f'_c$ for < 33% area of load application

   ($0.25\ f'_c$ = 750 psi; $0.375\ f'_c$ = 1125 psi)

13. Bearing Stress on Pool Floor for 4 12"-diameter Leveling Pads under each 10 x 7 Rack

σ_B = 322 psi for 1.2g vertically (SSE)

σ_B = 290 psi for 1.08g vertically (OBE)

σ_B = 269 psi for 1.0g (deadweight) (Normal)

14. Compressive Stress in Box Walls

σ_c = 780 psi for 0.2g horizontally, 1.2g vertically (SSE)

σ_c = 413 psi for 0.08g horizontally, 1.08g vertically (OBE)

σ_c = 169 psi for 0.0 horizontally, 1.0g vertically (Normal)

$0.6\ \sigma_y = 15{,}000$ psi

$0.6\ \sigma_{cr} = 7{,}105$ psi

VIII. RADIOLOGICAL EVALUATION

A. Direct Radiation

The principal source of radiation levels observed at the surface of the SFP water is due to the concentration of radionuclides within the pool water. This has been verified by calculations. The observed dose rate has been typically less than 5 mR/hr. The radionuclides are removed from the water by the SFP demineralizer with the need for changing the demineralizer resin determined by the pressure drop across the demineralizer. Increased fuel storage may result in an increased frequency of changing the demineralizer resin but is not expected to result in any increase in the radionuclide concentrations or in subsequent radiation levels at the surface of the water.

The top of the fuel assemblies stored in the spent fuel storage racks are at least 26 feet below the surface of the water. The 26 foot water shield reduces the direct radiation from the stored fuel assemblies to values which are negligible when compared to background.

In the original fuel racks, the sides of the fuel assemblies stored closest to the wall were approximately 12 inches from the concrete wall of the pool. The new fuel racks will reduce this distance to 11.3 inches. The slight reduction in distance and the closer fuel assembly spacing will result in a small increase in radiation levels outside the SFP. The resulting direct radiation levels outside the SFP wall will, however, remain below the design limits for the SFP wall.

Therefore, for the reasons mentioned, the increased fuel storage will have essentially no impact on the radiation levels at the surface of the water or outside the SFP walls.

B. Airborne Radioactivity

Increasing the storage capability of the SFP represents longer term storage of well cooled fuel. The additional spent fuel will have been stored for a year or more. The escape of gaseous or volative fission products from any defective fuel is expected to be negligible since most of the iodines and xenons have decayed after 100 days cooling time. Kr-85 remains relatively constant because of its long half-life. The thermal driving forces required to cause Kr-85 to diffuse from the defective fuel are not present; therefore, Kr-85 is expected to remain in the old fuel. Also, there is no method for particulate fission products to become airborne. Therefore, increased fuel storage will have essentially no impact on concentrations of radioactivity in the air of the auxiliary building.

IX. ACCIDENT ANALYSES

A. Fuel Handling Incident

The extent of damage that might result to a fuel assembly during fuel handling is addressed in the FSAR Section 14.2.1. The new rack is inherently stronger because of its box beam construction as compared to an open angle construction. Thus the above analyses are still valid.

B. Shipping Cask Drop Accident

The proposed spent fuel rack modification is entirely within the outer envelope of the existing spent fuel racks. The storage capability is increased by decreasing the spacing of the stored fuel rather than by rearrangement of the pool rack configuration.

Due to the current shortage of offsite spent fuel storage space and spent fuel reprocessing capability, the spent fuel cask will not be used for several years. An analysis of the "cask drop" accident will be submitted to the NRC prior to the use of a spent fuel cask.

C. Interruption of Spent Fuel Cooling

With the SFPCS in operation, the spent fuel heat load would be 5.3×10^6 Btu/hr with a 120°F pool temperature (Normal Refueling) and 9.3×10^6 Btu/hr with a 150°F pool temperature (Full Core Discharge). The volume of water in the SFP is approximately 255,000 gallons. Complete interruption of cooling would, therefore, result in maximum heatup rates for the pool of 2.5°F and 4.4°F per hour, respectively. The time for the pool to reach 180°F would be 24 hours starting from an initial pool temperature of 120°F in the Normal Refueling case and 6.8 hours starting from an initial pool temperature of 150°F in the Full Core Discharge case. In the time available, equipment maintenance can be accomplished or backup cooling can be obtained.

The system is designed to facilitate the installation of a portable pump if the SFP pump should be lost. In the event of loss of the SFPCS heat exchanger, cooling for the SFP can be provided by using temporary connections to one of the component cooling heat exchangers.

X. REFERENCES

1. R. F. Barry, "LEOPARD - - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094", WCAP-3269, September, 1963.

2. W. R. Cadwell, "PDQ-7 Reference Manual", WAPD-TM-678, January, 1967.

3. H. Bohl, E. Gelbard, and G. Ryan, "MUFT-4 - - Fast Neutron Spectrum Code for the IBM-704", WAPD-TM-72, July, 1957.

4. H. Amster and R. Suarez, "The Calculation of Thermal Constants Averaged over a Wigner-Wilkins Flux Spectrum: Description of the SOFOCATE Code", WAPD-TM-39, January, 1957.

5. L. E. Strawbridge and R. F. Barry, "Criticality Calculations for Uniform Water-Moderated Lattices", Nuclear Science and Engineering, 23, 58, 1965.

6. "Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period January 1 - March 31, 1965", WCAP-3269-12.

7. "List of Equipment and Apparatus at WREC", Westinghouse Reactor Evaluation Center, February, 1967.

8. W. L. Orr, H. I. Sternberg, P. Deramaix, R. H. Chastain, L. Bindler, and A. J. Impink, "Saxton Plutonium Program, Nuclear Design of the Saxton Partial Plutonium Core", WCAP-3385-51, December, 1965. (Also EURAEC-1940)

9. R. D. Leamer, W. L. Orr, R. L. Stover, E. G. Taylor, J. P. Tobin, and A. Bukmir, "PuO_2-UO_2 Fueled Critical Experiments", WCAP-3726-1, July, 1967.

10. A. F. Henry, "A Theoretical Method for Determining the Worth of Control Rods", WAPD-218, August, 1959.

11. P. W. Davison et al., "Yankee Critical Experiments - Measurements on Lattices of Stainless Steel Clad Slightly Enriched Uranium Dioxide Fuel Rods in Light Water," YAEC-94, Westinghouse Atomic Power Division (1959).

12. V. E. Grob and P. W. Davison et al., "Multi-Region Reactor Lattice Studies - Results of Critical Experiments in Loose Lattices of UO_2 Rods in H_2O," WCAP-1412, Westinghouse Atomic Power Division (1960).

13. W. J. Eich and W. P. Kovacik, "Reactivity and Neutron Flux Studies in Multiregion Loaded Cores," WCAP-1433, Westinghouse Atomic Power Division (1961).

14. W. J. Eich, Personal Communication (1963).

15. T. C. Engelder et al., "Measurement and Analysis of Uniform Lattices of Slightly Enriched UO_2 Moderated by D_2O-H_2O Mixtures,' BAW-1273, the Babcock & Wilcox Company (1963).

16. A. L. MacKinney and R. M. Ball, "Reactivity Measurements on Unperturbed, Slightly Enriched Uranium Dioxide Lattices," BAW-1199, the Babcock & Wilcox Company (1960).