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AUTH.NAME AUTHOR AFFILIATION
PARKER, W.O. Duke Power Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON, H.R. Office of Nuclear Reactor Regulation

SUBJECT: Forwards proposed amend to Tech Specs re increasing spent fuel storage capacity in spent fuel pool shared by Units 1 & 2.

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DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

February 2, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Robert W. Reid, Chief
Operating Reactor Branch #4

Re: Oconee Nuclear Station
Docket Nos. 50-269, 270

Dear Sir:

Pursuant to 10CFR 50, §50.90 please find attached a proposed amendment to the Oconee Nuclear Station Technical Specifications which concerns increasing the spent fuel storage capacity in the spent fuel pool shared by Units 1 and 2. The proposed amendment would allow the expansion of the pool capacity from the currently licensed limit of 336 assemblies to 750 assemblies. The proposed specification, Attachment 1, limits the enrichment of the fuel which is to be stored in the pool, the number of spent fuel locations, and the decay time of fuel in the vicinity of the cask handling area of the pool. In support of this, a report is also attached which describes the methods by which the spent fuel capacity will be increased as well as providing a safety evaluation and environmental appraisal for the proposed action. The guidance contained in the staff document "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" and Federal Register Notice published September 10, 1973 (FR 42801) has been utilized in the preparation of this document.

In my letter of October 18, 1978 to Mr. R. W. Starostecki, Chief, Fuel Reprocessing and Recycle Branch (NRC/ONMSS) and subsequent conversations with Mr. Starostecki and members of the staff, Duke presented a preliminary conclusion with regard to the feasibility of pursuing expansion of the Oconee 1 and 2 spent fuel pool. At that time, it was assumed that a modification to the spent fuel cooling system would have to be installed concurrently with reracking the pool. Since that time, further evaluation has determined that reracking could proceed prior to completion of the cooling system modification. Furthermore, it has been determined that it may be possible to expedite other facets of design/fabrication/installation sufficiently to complete the reracking between the Unit 3 refueling, anticipated to start in mid-May, 1979 and the Unit 1 refueling presently scheduled for November, 1979. If licensing delays do not extend beyond the June time frame requested reracking can proceed without necessitating shipment of spent fuel off site. If significant delays occur it may be necessary to ship fuel off site before the rerack can be completed.

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$ 12,300.00

The Atomic Safety Licensing Board (ASLB) is currently reviewing another proposal which will also serve to alleviate the shortage of storage space at Oconee Nuclear Station. This proposal which concerns authorization to store spent fuel from Oconee at McGuire Nuclear Station has not yet been approved. This action has been contested by three intervenors and the hearing date has not yet been set. Therefore, it has been deemed prudent to pursue the additional action which can preclude, in the short term, the totally unacceptable alternative of shutting down one or more of the Oconee units.

It should be emphasized that neither transshipment or reracking were originally contemplated, however changes in national policy now necessitate these options. Furthermore, of the two options presented, the transshipment option is considered to be more environmentally benign and economically justified. However, the review of the options will be carried out separately, in that the transshipment option is being dealt with as a 10CFR 70 license amendment for McGuire Nuclear Station. Additionally, as indicated in Section 8 of the attached document the utility of the two options are independent of each other. Approval of the transshipment option will promote ALARA exposure levels during the reracking and approval of reracking will increase flexibility with regard to utilization of Away-From-Reactor storage sites which might be available in addition to McGuire.

It is therefore requested that review of this request proceed as expeditiously as possible and that approval be granted no later than June, 1979 to accommodate an already limited schedule. Delays which extend the review beyond the requested approval date could jeopardize the timely installation of the modification during 1979.

This proposal is considered to constitute a single Class IV license amendment request. As such, please find attached a check in the amount of $12,300 to cover the required license fees associated with the review of this document.

Very truly yours,

William O. Parker, Jr.

KRW:scs
Attachment

WILLIAM O. PARKER, JR., being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this request for amendment of the Oconee Nuclear Station Technical Specifications, Appendix A to Facility Operating Licenses DPR-38, DPR-47 and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

William O. Parker, Jr.

William O. Parker, Jr., Vice President

Subscribed and sworn to before me this 1st day of February, 1979.

Vivian B. Robbins

Notary Public

My Commission Expires:

February 15, 1982

50-269/270
ltr 2-2-79
7902080228

ATTACHMENT 1

PROPOSED

TECHNICAL SPECIFICATION REVISION

3.8 FUEL LOADING AND REFUELING

Applicability

Applies to fuel loading and refueling operations.

Objective

To assure that fuel loading and refueling operations are performed in a responsible manner.

Specification

3.8.1 Radiation levels in the reactor building refueling area shall be monitored by RIA-2 and RIA-3. Radiation levels in the spent fuel storage area shall be monitored by RIA-6. If any of these instruments becomes inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

3.8.3 At least one low pressure injection pump and cooler shall be operable.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required to shutdown the core to a $k_{eff} \leq 0.99$ if all control rods were removed.

3.8.5 Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place.

3.8.6 During the handling of irradiated fuel in the reactor building at least one door on the personnel and emergency hatches shall be closed. The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

3.8.7 Both isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed

3.8.8 When two irradiated fuel assemblies are being handled simultaneously within the fuel transfer canal, a minumum of 10 feet separation shall be maintained between the assemblies at all times.

Irradiated fuel assemblies may be handled with the Auxiliary Hoist provided no other irradiated fuel assembly is being handled in the fuel transfer canal.

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

3.8.10 The reactor building purge system, including the radiation monitor, RIA-45, which initiates purge isolation, shall be tested and verified to be operable immediately prior to refueling operations.

3.8.11 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours.

3.8.12 Two trains of spent fuel pool ventilation shall be operable with the following exceptions:

a. With one train of spent fuel pool ventilation inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the operable spent fuel pool ventilation train is in operation and discharging through the Reactor Building purge filters.

b. With no spent fuel pool ventilation filter operable, suspend all operations involving movement of fuel within the storage pool or crane operations with loads over the storage pool until at least one train of spent fuel pool ventilation is restored to operable status.

3.8.13 a. Prior to spent fuel cask movement in the Unit 1 and 2 spent fuel pool, spent fuel stored in the first 28 rows of the pool closest to the spent fuel cask handling area shall be decayed a minimum of 50 days.

b. Prior to spent fuel cask movement in the Unit 3 spent fuel pool, spent fuel stored in the first 20 rows of the pool closest to the spent fuel cask handling area shall be decayed a minimum of 43 days.

3.8.14 No fuel will be stored in either spent fuel pool which has an enrichment greater than 3.5 weight percent U^{235}.

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.7 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The low pressure injection pump is used to maintain a uniform boron concentration. (1) The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the

core. (2) The boron concentration will be maintained above 1,800 ppm. Although this concentration is sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The k_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing of the Reactor Building purge isolation is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

The off-site doses for the fuel handling accident are within the guidelines of 10 CFR 100; however, to further reduce the doses resulting from this accident, it is required that the spent fuel pool ventilation system be operable whenever the possibility of a fuel handling accident could exist.

Specification 3.18.13 is required as the safety analysis for a postulated cask handling accident was based on the assumptions that spent fuel stored as indicated has decayed for the amount of time specified for each spent fuel pool.

REFERENCES

(1) FSAR, Section 9.7

(2) FSAR, Section 14.2.2.1

(3) FSAR, Section 14.2.2.1.2

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

Specification

5.4.1 New Fuel Storage

5.3.1.1 New fuel will normally be stored in the spent fuel pool serving the respective unit.

In the spent fuel pool serving Units 1 and 2, the fuel assemblies are stored in racks in parallel rows, having a nominal center-to-center distance of 13.75 inches in both directions. This spacing is sufficient to maintain a $K_{eff} \leq 0.95$ when flooded with unborated water, based on fuel with an enrichment of 3.5 weight percent U^{235}.

In the spent fuel pool serving Unit 3, the fuel assemblies are stored in racks consisting of stainless steel cavities which maintain a minimum edge-to-edge spacing of 3.95 inches between adjacent fuel assemblies. The neutron poisoning effect of the storage cavity material combined with the minimum 3.95 inches edge-to-edge spacing between adjacent fuel assemblies is sufficient to maintain a $K_{eff} \leq 0.95$ when flooded with unborated water based on fuel with an enrichment of 3.5 weight percent U^{235} or the equivalent.

5.4.1.2 New fuel may also be stored in the fuel transfer canal. The fuel assemblies are stored in five racks in a row having a nominal center-to-center distance of 2' 1-3/4". One rack is oversized to receive a failed fuel assembly container. The other four racks are normal size and are capable of receiving new fuel assemblies.

5.4.1.3 New fuel may also be stored in shipping containers.

5.4.1.4 New fuel of enrichment not exceeding 2.9 weight percent U^{235} or the equivalent may be placed in dry storage in Unit 3 fuel storage racks in a checkerboard pattern, with fuel assemblies occupying only diagonally adjacent storage locations. Unused storage locations in a fuel storage module shall be covered by inserting a metal plate in the lead-in to prevent incorrect placement of fuel assemblies. This configuration is sufficient to assure a $K_{eff} \leq 0.9$ at all times.

5.4.2.1 Irradiated fuel assemblies will be stored, prior to offsite shipment, in a stainless steel lined spent fuel pool.

The spent fuel pool serving Units 1 and 2 is sized to accommodate a full core of irradiated fuel assemblies in addition to the concurrent storage of the largest quantity of new and spent fuel assemblies predicted by the fuel management program.

Provisions are made in the Unit 1, 2 spent fuel pool to accomodate up to 750 fuel assemblies and in the Unit 3 spent fuel pool up to 474 fuel assemblies.

5.4.2.2 Spent fuel may also be stored in storage racks in the fuel transfer canal when the canal is at refueling level.

5.4.3 Except as provided in Specification 5.4.1.4, whenever there is fuel in the pool, the spent fuel pool is filled with water borated to the concentration that is used in the reactor cavity and fuel transfer canal during refueling operations.

5.4.4 The spent fuel pool and fuel transfer canal racks are designed for an earthquake force of 0.1g ground motion.

REFERENCES

FSAR, Section 9.7

ATTACHMENT 2

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 1 and 2

INFORMATION IN SUPPORT of

SPENT FUEL POOL MODIFICATION

February 2, 1979

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1.0 INTRODUCTION

The Oconee Nuclear Station was designed and constructed with two spent fuel storage pools-one associated with Units 1 and 2 and one with Unit 3. The design was such that the pools would be capable of storing 1 2/3 and 1 1/3 cores respectively. Both the Oconee Nuclear Station Final Safety Analysis Report and Technical Specifications address the adequacy of this to "accommodate a full core of irradiated fuel assemblies in addition to concurrent storage of the largest quantity of new and spent fuel assemblies predicted by the fuel management program" (References 1 and 2). The actual designed capacity for each pool was 336 and 216 locations.

In 1975 it was deemed prudent to increase the storage capacity at the Oconee site. The Unit 1 and 2 pool contained spent fuel from the initial Unit 1 refueling in 1974. The Unit 3 pool was empty of any spent fuel. Thus, it was decided to increase the capacity of the Unit 3 pool. A request to amend the Unit 3 Operating License, DPR-55, was submitted on September 12, 1975 and was approved, as Amendment No. 17, on December 22, 1975. The completed modification increased its capacity to 474 locations (including failed fuel). It was considered at that time that the resulting combined on-site capacity (810 locations) would be sufficient to store spent fuel until such time as shipment to the Allied General Nuclear Services reprocessing plant could begin. The modification of the Unit 1 and 2 pool in this time frame would have had to have been done "wet" (with spent fuel present in pool). Such operations were considered to exceed state-of-the-art technical capabilities at that time.

On April 7, 1977, President Carter issued a policy statement on commercial reprocessing of spent nuclear fuel which effectively eliminated reprocessing as an end to the nuclear fuel cycle, at least in the near future. On October 18, 1977, the Department of Energy accepted ultimate responsibility for storing spent nuclear fuel. On December 23, 1977, the GESMO proceedings were deferred indefinitely. The combined effect of this national policy was to leave operating nuclear plants, like Oconee, without a repository for the spent fuel previously generated or being generated, other than expanded storage provided by the owner/operator. Two plausible alternatives are left for Oconee: (a) ship the spent fuel to other Company nuclear plants; or (b) expand the Unit 1 and 2 pool. Both options, along with several others, have been considered and are presented in Section 7 of this report. By letter dated March 9, 1978, Duke Power Company requested approval of option (a) using McGuire Nuclear Station as a temporary storage site for some Oconee fuel until another site can be identified or constructed. By the letter to which this report is attached, Duke is requesting approval for option (b).

The proposed expansion of the Unit 1 and 2 pool capacity will, if approved, allow the storage of up to 750 assemblies in that pool and 1224 on-site (including "failed-fuel" locations). The proposed modification will cause the temporary loss of full core reserve capacity during some phase(s) of the modification but will allow the retention of that capability for an extended period. Table 1.0-1 shows the scheduled refuelings for the Oconee Nuclear Station which can be accommodated by the modified fuel racks. In order to maintain a full core reserve, at least 177 locations must remain available. The difference between the total number of licensed locations and the number of "useable" locations is due to storage of irradiated components in the pool, the presence of Post-Irradiation Examination (PIE) equipment and an interference from structures

in the pool area. It should be noted that the modification, as proposed, must be completed prior to the first refueling in the fourth quarter of 1979 due to space requirements and ALARA considerations. Both options (a) and (b) as described above will be utilized to prevent the unacceptable alternative of shutting down Oconee entirely at some, relatively near, point in time. Neither alternative is a permanent solution but both have been reviewed in generic and specific cases (References 3 and 4) and found to be acceptable.

The following chapters are provided with intent to provide information necessary for review and approval of the request for amendment to the Oconee Nuclear Station Technical Specifications (Attachment 1). Sections 2 through 5 provide sufficient information in order to support issuance of a Safety Evaluation Report (SER) on the subject. Sections 6 through 8 should provide adequate support for issuance of an Environmental Impact Appraisal in conjunction with a Negative Declaration on the proposed modification. It is considered that the modification is not inimical to the health and safety of personnel or the general public and that it represents an environmentally acceptable alternative which meets the requirements of NEPA and the guidance provided by the Commission on such applications (References 5 and 6).

References

1. Oconee Nuclear Station Technical Specification Section 5.4.2.1
2. Oconee Nuclear Station Final Safety Analysis Report Section 9.7.1.3
3. NUREG-0404 Generic Environmental Impact Statement on Handling and Storage of Spent LWR Fuel
4. Carolina Power & Light, Brunswick/Robinson Transshipment (Amendment 8 to DPR-71 and Amendment 30 to DPR-62 approved by letter dated August 26, 1977)
5. "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications"
6. September 16, 1975 Federal Register Notice (FR-42801)

TABLE 1.0-1
OCONEE NUCLEAR STATION

Anticipated Refueling Schedule/Storage Space Inventory

			Without Reracking Or Transshipment		With Reracking Only
Unit	Estimated Refueling Date	Anticipated No. Of Assemblies Discharged	Total No. Of Rack Locations Remaining After Refueling	No. Of Useable Rack Locations Remaining After Refueling	Total No. Of Rack Locations Remaining After Refueling
Current Status		545 (current)	265 (current)	214 (current)	---
3	May, 1979	56	209	158	623
1	November, 1979	68	141	90	555
2	January, 1980	68	73	22	487
3	May, 1980	56	17	---	431
1	April, 1981	72	---	---	359
3	May, 1981	56	---	---	303
2	June, 1981	72	---	---	231
3	May, 1982	56	---	---	175
1	September, 1982	72	---	---	103
2	January, 1983	72	---	---	31
3	July, 1983	56	---	---	---

2.0 RACK DESIGN

2.1 DESIGN BASES

The proposed modification provides for accommodating up to 744 fuel assemblies in the Unit 1 and 2 Spent Fuel Pool and consists of replacing the existing fuel assembly storage racks with the Combustion Engineering, Inc., supplied High Capacity (Hi-Cap) Fuel Assembly Rack without changing the basic structural geometry of the spent fuel pool as described in FSAR Section 9.7. In addition to the 744 spent fuel locations, 6 failed fuel cavities are also included.

The Hi-Cap fuel assembly storage rack modules consist of an array of storage cavities with nominal center-to-center spacing of 13.75 inches; each storage cavity can accommodate one fuel assembly. Fourteen independent fuel assembly storage modules are to be installed in the pool. Each storage module is made up of square storage tubes joined together by a chevron grid structure with each tube capable of accepting one fuel assembly. The storage tubes have lead-in surfaces in top castings to provide guidance for insertion of fuel assemblies. The chevron grid structure assures that the required minimum storage tube spacing is maintained for all design conditions, including the safe shutdown earthquake. The fourteen modules are not interconnected. Each module rests independently on the liner plate of the pool floor. Cooling of the spent fuel assemblies is achieved or forced through natural or forced circulation.

The design provides storage capacity for fuel assemblies having a fuel enrichment of 3.5 weight percent U-235 in UO_2 or the equivalent. The fuel is maintained in a safe and subcritical configuration during normal and abnormal conditions. The effective multiplication factor (K_{eff}) shall not exceed 0.95 when all uncertainties are included in the K_{eff} analysis. Postulated accident conditions do not increase the K_{eff} value to greater than 0.95 when all uncertainties are taken into account.

The Duke Power Company is responsible for the modification and implementation thereof. Combustion Engineering, Incorporated, will design and manufacture the racks and provide installation technical assistance.

2.2 DESIGN DESCRIPTION

The Hi-Cap fuel assembly storage rack is designed to maintain the stored fuel, having a fuel enrichment of 3.5 weight percent U-235 in UO_2 or equivalent, in a safe, coolable, and subcritical configuration during normal and abnormal conditions.

The completed storage rack envelope is a rectangular array composed of fourteen modules with twelve modules consisting of a 6 x 9 array for spent fuel storage and two modules consisting of a 6 x 8 array for spent fuel storage with a 1 x 3 array attached for failed fuel storage. The fourteen modules are arranged as shown in Figure 2.2-1.

Each storage rack module will be self-supporting and will rest on base plates, placed on the pool floor. The interface with pool boundaries and between modules will be designed to transfer normal and shear loads to the pool liner and bottom slab.

Each fuel assembly storage module is composed of rectangular storage cavities fabricated from one-quarter inch thick stainless steel plate, with each cavity capable of accepting one fuel assembly. The fuel assembly storage cavities have lead-in surfaces at the top to provide guidance for insertion of fuel assemblies. The cavities are open at the top and bottom to provide a flow path for convective cooling of the fuel assemblies through natural circulation. The fuel assembly storage cavities are connected by a chevron grid structure to form modules which limit structural deformations and maintain a nominal center-to-center spacing of 13.75 inches between adjacent storage cavities during design loading conditions including seismic. The design and configuration of a typical 6 x 9 fuel assembly storage module is shown in Figure 2.2-2.

Reference design codes, specifications, standards, NRC Regulatory Guides, etc. applicable to the fuel racks are as follows.

(a) Regulatory Guide 1.13, "Fuel Storage Facility Design Basis".

(b) Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants".

(c) Regulatory Guide 1.29, "Seismic Design Classifications".

(d) Standard Review Plan, Section 9.1.2, "Spent Fuel Storage".

(e) ASME Boiler and Pressure Vessel Code, Section III, Subsection NF.

(f) ASME Boiler and Pressure Vessel Code, Section IX.

(g) Project Specification for Hi-Cap Storage Racks 16678-RCE-0451

(h) Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants".

(i) Regulatory Guide 1.61, "Damping Valves for Seismic Design of Nuclear Power Plants".

(j) Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis".

2.3 DESIGN EVALUATION

2.3.1 STRUCTURAL AND SEISMIC ANALYSIS

Seismic analyses of the fuel storage racks are performed by the nonlinear time history method for each of the two horizontal directions and by the response spectrum method for the vertical direction. The time histories and response spectrum utilized in these analyses represent the responses of the pool structure to the specified ground motion.

Reference is made to Project 81 PSAR, Docket Nos. STN50-488 through -493, Section 3.7. The smoothed response spectra shown on Figure 2E-2A were normalized to 0.10g for Safe Shutdown Earthquake (SSE). The normalized seismic input spectra conform to the requirements of Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants." An earthquake acceleration-time history compatible with these spectra, as shown in Figures 2E-2B through 2E-2E, was used as a base motion on the model of the Auxiliary Building.

The seismic response of the Auxiliary Building to the base excitation is determined by a dynamic analysis. The dynamic analysis is made by idealizing the structure as a series of lumped masses with weightless elastic columns acting as spring restraints. The base of the structure is considered fixed. The choice of the location of the mass-joints depends on the distribution of masses in the real structure.

The modes and frequencies of natural vibration are obtained. A time-history analysis is performed using the modal superposition method, in which the responses in the normal modes are determined separately, then superimposed to provide the total response. The structural damping values used are 4 percent for ½ SSE and 7 percent for SSE. These values conform to the requirements of Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

The acceleration-time history response of the fuel pool floor mass-joint was then used a input for the time history analysis of the fuel rack model. The structural damping value used in the seismic analysis of the racks if 4% for the rack structure. This value is consistent with the damping values specified in Regulatory Guide 1.61. The combination of modal responses for the vertical rack analysis and combination of direction responses for horizontal and vertical analyses are in accordance with Regulatory Guide 1.92.

Typical mathematical models used to determine the horizontal response loads are shown in Figs. 2.3.1-1 and 2.3.1-2. A modal extraction analysis of a space frame model (see Fig. 2.3.1-1) of the empty rack module in air is performed with the SAP (Reference 1) computer code. The modal parameters of the rack module are then used to derive a dynamically equivalent model of the module which is then incorporated into a lateral nonlinear model (see Fig. 2.3.1-2) that includes both the rack and contained fuel assemblies, and the water surrounding and contained within the cavities. The analysis is performed with CESHOCK, a proprietary version of the SHOCK (Reference 2) computer code, and the results include the impact force time histories between the fuel assemblies and cavities and the rack/pool interface reaction loads.

The effects of gaps and submergence in water on the motion of the fuel racks are accounted for in the seismic response analysis. This is necessary because the fuel which responds to seismic excitation at its own natural frequency, will move freely through the available gap and impact the storage cavity. Also, as the fuel moves within the rack and the rack moves relative to the pool, the water between these structures is moved or accelerated by them. This acceleration of the water introduces hydraulic loads on these structures and results in a lowering of natural frequencies of the fuel and rack. The hydraulic forces are accentuated when the interacting submerged

structures are in close proximity (small gaps). These interactive effects of the motion of the adjacent structures (pool, racks and fuel) on each other through the surrounding and contained water are referred to as hydrodynamic coupling. Hydrodynamic coupling is expressed in terms of forces proportional to the relative accelerations between the structures. The coefficients of these acceleration terms are the elements of the hydrodynamic mass matrix for the mathematical model, and are functions of geometry and proximity of the structures. The hydrodynamic mass method of analysis of structures immersed in a fluid is presented in References 3-5.

The load transfer to the pool structure from the fuel racks occurs only at the base of the racks and consists of the vertical compression loading and horizontal shear forces due to frictional restraint. Two limiting friction conditions between the racks and pool floor are considered in the horizontal seismic analysis of the racks. For the first case, an infinite friction coefficient is assumed since this provides the peak structural loading of the rack members and pool interfaces. Next, the motion of the racks relative to the pool floor is determined to assure that the spacing between the racks and pool is sufficiently large so as to preclude contact. This is done using low values of friction. Coefficients of friction utilized in the analysis are based on test data provided by Reference 6 for stainless steel in water (0.4 to 2.0 at a velocity of 3 mm/sec). Its minimum value (0.4) is extrapolated to a static value (0.55) and a dynamic value (0.28) at a velocity typical of the analysis. The extrapolation is based upon data provided by Reference 7 indicating the effect on coefficient of friction of the relative velocity.

Normal operating loads include dead weight and thermal expansion loads.

The following loads are considered in the design of the fuel racks.

(a) Normal Loads

Normal loads are due to the fuel rack and supporting structure. These loads are determined assuming all cavities contain a fuel assembly and include the dead weight of the fuel rack and supporting structure.

(b) Thermal Loads

Thermal loads resulting from the following conditions are considered:

(b.1) Pool bulk temperature excursion from 70° F to 150° F.
(b.2) Pool bulk temperature excursion from 70° F to 250° F (faulted).
(b.3) Temperature difference between adjacent cavities is assumed to be 20°F.

(c) Seismic Loads

Seismic loadings on the racks and the embedments, along with maximum deflections, are obtained for maximum earthquake (E') conditions with a time history analysis using the appropriate time histories. The maximum fuel assembly impact loadings are also obtained from this analysis.

(d) Shipping and Installation Loads

Loads on the rack due to its own weight are determined for shipping, upending and installation operations.

(e) Dropped Fuel Bundle Loads

The design drop height is six feet to the point of impact on the cavity lead-in funnel. The total kinetic energy of the 1,700 lb. fuel bundle at the point of impact is 122,400 in-lb. The impact energy is transmitted through the lead-in funnel to be absorbed by the upper grid and adjacent structure.

The above loads are combined as follows to determine the maximum stress condition, and do not exceed the design allowable stress limits stated in Table 2.3-1.

Normal Conditions

1. The loads from paragraph (d) acting separately.

2. The loads from paragraphs (a), (b.1), (b.3) acting simultaneously.

Faulted Conditions

1. The loads from paragraphs (a) and (b.2) acting simultaneously.

2. The loads from paragraphs (a), (b.1) and a maximum earthquake (E') acting simultaneously.

The maximum uplift load available from the fuel handling crane is a less severe loading on the fuel rack structure than that resulting from the maximum earthquake (E').

Structural design precludes placing a fuel assembly between cells, and the rack will withstand the loadings imposed by a postulated dropped fuel assembly.

<u>REFERENCES</u>

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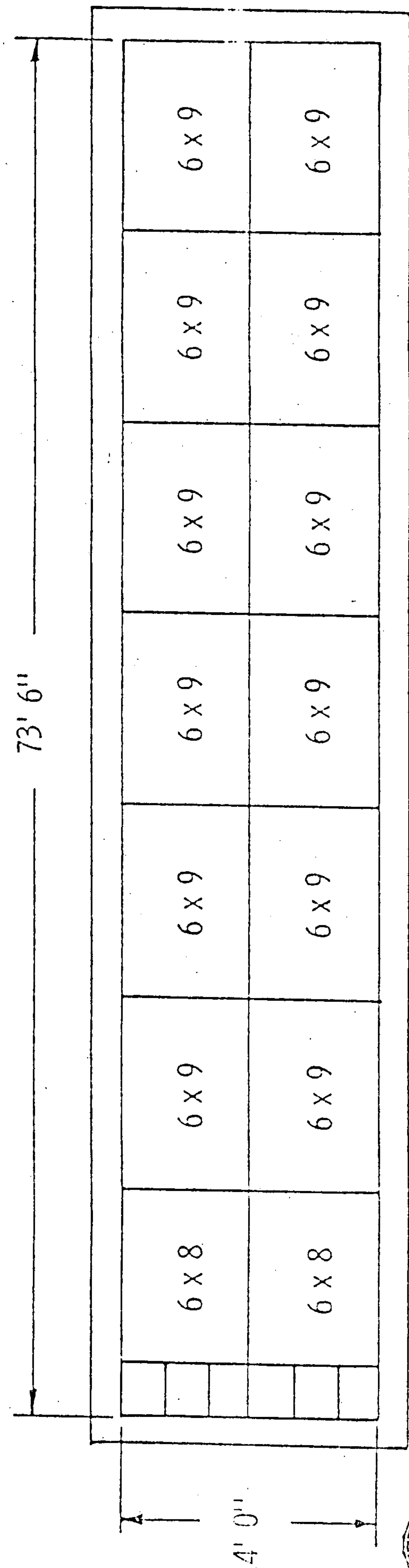
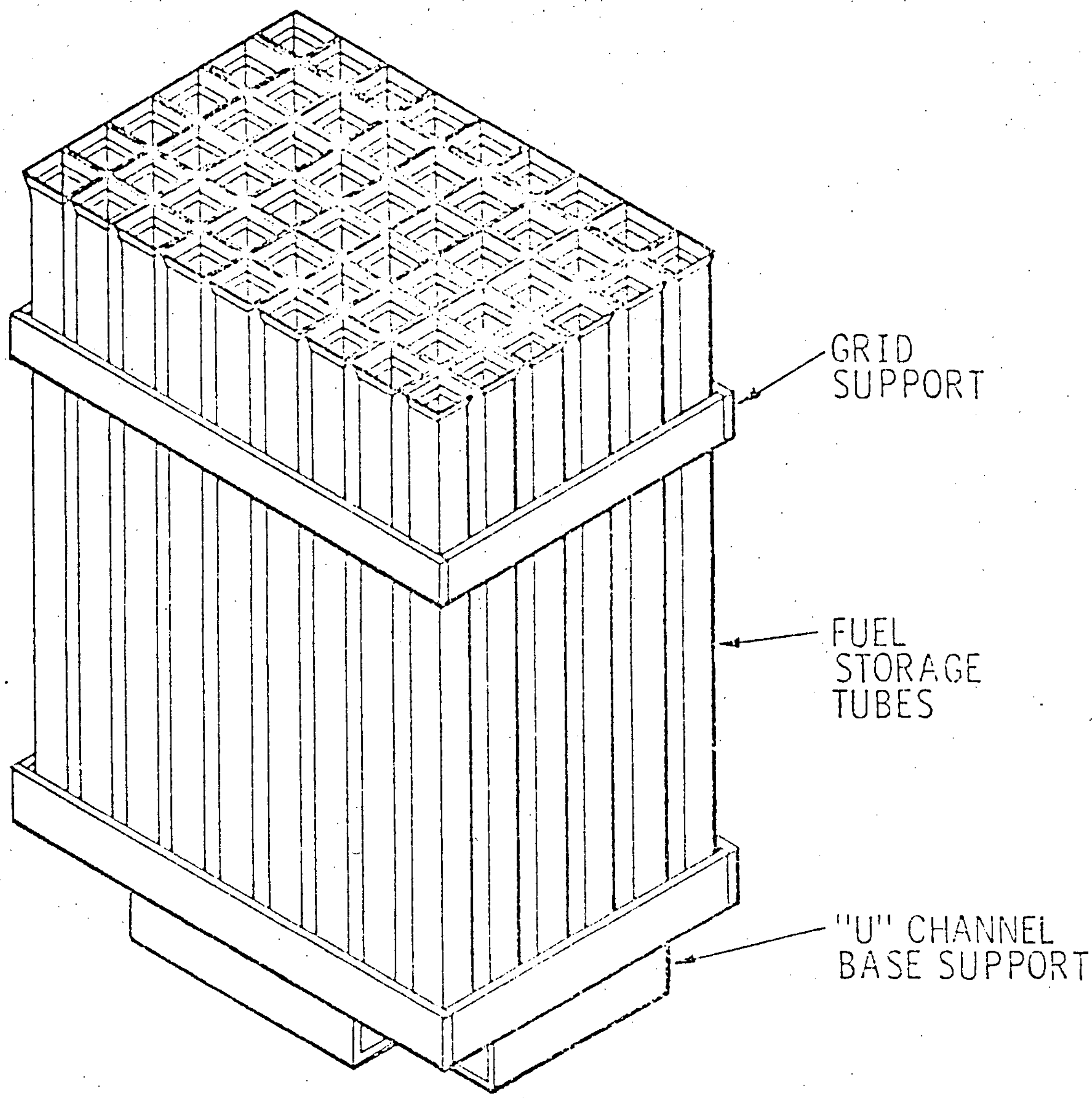


HI-CAP SPENT FUEL RACK LAYOUT
OCONEE NUCLEAR STATION
Figure 2.2-1



FREE STANDING SPENT FUEL MODULE (6X9)
OCONEE NUCLEAR STATION
Figure 2.2-2

2.3.2 NUCLEAR CRITICALITY ANALYSIS

Analysis is performed to insure that the multiplication factor of the storage rack, including calculational uncertainties, is less than 0.95.

The criticality analysis for the spent fuel storage pool is carried out assuming: (1) an infinite array of fuel storage locations, (2) each storage location consists of square type 304 stainless steel box containing an unirradiated UO_2 fuel assembly having a fuel enrichment of 3.5 weight percent U-235, (3) the fuel storage pool is flooded with pure (unborated) water at a temperature of 68° F, and (4) no control rods are stored in the fuel assemblies. The effective multiplication factor is determined for the nominal fuel rack design dimensions, i.e., with nominal box lattice pitch, box wall thickness, and fuel assembly placement. Analyses are also carried out for an infinite array cluster of four fuel storage locations assuming: (1) the most adverse combination of fuel rack dimensional tolerances and (2) a subsequent displacement of the fuel assemblies into their most reactive positions (closest approach of 4 neighboring fuel assemblies). The increase in multiplication factor, relative to nominal conditions, is viewed as a conservative measure of the effect of fabrication tolerances and fuel displacement since it assumes worst conditions throughout the whole fuel rack rather than the statistical distribution of such effects which would be expected to occur in the fabrication and fuel loading processes. The analyses described above contain additional conservatism in the sense that: (1) reactivity losses due to axial leakage from the active fuel volume as well as parasitic absorptions in fuel pin spacer grids are not included in the above multiplication factors, and (2) the parasitic contribution of structural members other than the box walls are not included in the quoted multiplication factors.

An overall calculational uncertainty is determined and assigned to the criticality analysis. This uncertainty is composed of two components - a calculational uncertainity, and an uncertainty for the thickness of the stainless steel plates. The calculational uncertainty is deduced from the analysis of reactor and laboratory experiments. There is a 95% probability that at least 95% of the measured multiplication factors will be less than the calculated value plus the calculated uncertainty.

The reactivity effects of the stainless steel structure in the spent fuel storage racks are included in the criticality analyses in a conservative manner by virtue of the following assumptions and/or design procedures.

(a) No reactivity credit is taken in the criticality analyses for stainless steel structure (e.g., shear plates, spacer channels, etc.) other than the box wall structure.

(b) The specified nominal box wall thickness and tolerances for the spent fuel rack are 0.250 (+0.045, -0.010) in accordance with ASTM-240.

Thus in criticality analyses for the so called nominal and most adverse combination of fabrication tolerances, values for the box wall thickness are taken to be 0.250 and 0.240 inches, respectively. Since the box wall thicknesses encompassed by the specified tolerances fall in a range for which the derivative of steel worth with thickness is positive (-0.11 Δ K/in.), increases in plate thickness beyond the minimum will result in lower multiplication factors for the spent fuel rack.

The modeling and computer codes are described below:

CEPAK

The CEPAK lattice program is employed to calculate the basic broad group cross section data for the fuel assembly, spent fuel rack structure, and water. This program is a synthesis of a number of computer codes, many of which were developed at other laboratories, e.g., FORM (Reference 1), THERMOS (Reference 2) and CINDER (Reference 3). These codes are interlinked in a consistent way with inputs from an extensive library of differential cross section data.

NUTEST

NUTEST is a two-dimensional integral transport code which employs the collision probability technique to compute sub-region dependent reaction rates in an explicit geometric representation of the fuel rods and associated structure of a fuel assembly. This code is used to calculate the flux advantage factors which are applied as correction factors to the basic broad group cross sections computed by the CEPAK lattice program to account for heterogeneous lattice effects not represented in either the multigroup spectrum or homogenized cell spatial calculation, e.g., heterogeneous fast fission effect in fuel pellets.

DOT-2W

The spatial flux solution and multiplication factor for an infinite array of individual or clusters of fuel storage cells are computed with the two dimensional, discrete ordinates transport code, DOT-2W (Reference 4).

Qualification of the calculational model and evaluation of calculation uncertainties and/or bias factors are based on the analysis of a variety of reactor and laboratory experiments. The results of the analysis of a series of UO_2 critical experiments are summarized in Table 2.3-2. To supplement this group of critical experiments, the following results of analyses at the cold, zero power, beginning of life conditions for five reactor lattices are provided.

REACTOR	TEMP. (^oF)	SOLUBLE BORON (PPM)	K_{eff}
San Onofre No. 1	163	2254	1.0002
Maine-Yankee	260	955	0.9976
Fort Calhoun	260	900	0.9986
Calvert Cliffs No.1	260	1048	1.0010
Stade - 1	221	1302	1.0049

Although the spatial solution for the flux distribution was obtained by use of diffusion theory code such as PDO-7 (Reference 5), transport corrections for the reflector and heterogeneous lattice effects were employed.

Since fuel storage arrays do involve the spacing of the fuel assemblies at larger separation distances than in typical PWR reactor lattices, the predictive capability of the calculational model was tested on the following experiments. In these analyses, the spatial flux solution was obtained directly with the transport code, DOT-2W. To assess the accuracy of the calculational model in predicting the multiplication factor of fuel assemblies having a separation distance

sufficiently large so as to be isolated, analyses were carried out for a group of subcritical exponential experiments on clusters of 3.0 w/o UO_2 fuel pins clad with Type 304 stainless steel and moderated by H_2O (page 165 of Reference 6). The cluster sizes analyzed vary from 181 to 301 fuel rods to encompass the range of sizes typical of current PWR fuel assemblies. The multiplication factors for the lattices analyzed with DOT-2W using axial bucklings deduced from the reported relaxation lengths are tabulated below.

No. of Fuel Rods	K_{eff}
181	0.999
211	1.004
235	1.000
265	1.002
301	1.002

These results indicate that the calculational model predicts the multiplication factor for small clusters of fuel rods in a water environment to a high degree of accuracy.

As a further verification of the calculational model, an analysis was carried out for the ten fuel assembly critical configuration measured in the LACBWR startup criticals and reported in Reference 7. The LACBWR fuel assembly consists of a 10 x 10 array of Type 304 stainless steel clad, 3.63 w/o enriched UO_2 fuel rods enclosed in a 0.078 inch thick shroud. The ten fuel assemblies in the critical configuration employed stainless steel shrouds; the water gap between adjacent fuel assembly shrouds was 0.75 inches. The calculational model when applied to the ten assembly configuration yielded a multiplication factor of 0.015 versus the experimental value of 0.01 reported in Reference 7, thus the calculational model yields a conservative estimate of the multiplication factor for this experimental configuration.

A calculational uncertainty of 0.006 was deduced at the 95/95 confidence level using the differences between calculated and measured multiplication factors for the complete series of UO_2 experiments described above.

The magnitude of the uncertainty allowance for the one quarter inch thick stainless steel plates was derived from an analysis of the worth of steel water reflectors on the criticality of a low enrichment fuel region. To benchmark this analysis, selected portions of the ORNL experiments reported in Reference 8 were analyzed. In these experiments, the critical height of the 4.98 weight percent enriched uranyl floride solution was measured for thicknesses of Stainless Steel Type 304 of up to 5.08 cm inserted between the 0.079 cm thick solution container and the radial, infinitely thick water reflector. To assess the accuracy of the calculational model in predicting the worth of stainless steel inserted between a fuel and water reflector region, the following calculations were carried out with the DOT-2W code using an RZ representation for the experimental configuration employing the 33.02 cm diameter solution container. Multiplication factors were calculated for the critical solution heights corresponding to the zero and 0.64 cm thick reflector steel thicknesses; in addition, the worth of the 0.64 cm thick steel insert was determined by calculating the increase in multiplication factor when the steel was removed by the critical solution height was unchanged. In these analyses, the multiplication factor for the 0.64 cm steel case was less than that for the no steel condition by 0.0041 and

the worth of the steel was calculated to be 0.0239. This information combined with the calculated worth of a one quarter inch thick stainless steel box surrounded an isolated fully reflected 3.5 w/o enrich, B & W 15 x 15 fuel assembly implies a calculational bias for the steel of 0.012 Δ k.

The reactivity balance for the criticality analysis of the normal spent fuel storage locations is summarized as follows:

Design Limit	0.950
Calculational Uncertainty	-0.006
Stainless Steel Calculational Bias	-0.012
Design Target	0.932

	Design Conditions	
	Nominal	Most Adverse
Multiplication Factor for Spent Fuel Storage Rack	0.901	0.929
Excess Margin	0.031	0.003

The nuclear criticality analysis was performed for a uniform pool temperature of 68^{o} F. To assess the multiplication factor change with temperature, an additional calculation at 150^{o} F was performed. The calculation showed a decrease in reactivity (from 68^{o} F) of about 0.001 Δk. For this analysis, all materials and dimensions (except for the fuel pellet) were expanded from the 68^{o} F condition to 150^{o} F. Nuclear cross-sections at 150°F were also employed.

2.3.3 THERMAL-HYDRAULIC ANALYSIS

An analysis of the maximum fuel cladding temperature is performed for the postulated case of complete loss of coolant circulation to the pool. Natural circulation flow rates within the storage tubes of similar racks have been calculated which give confidence that convection film co-efficients in excess of 50 BTU/hr-ft^2 °F can be expected for the Oconee 1 and 2 racks. Because the heat flux is small, very large uncertainties in the film coefficient are acceptable without causing probibitively high clad temperatures. The fuel cladding temperature analysis verifies that the temperature of the clad in the spent fuel pit is maintained at acceptable levels for the Oconee 1 and 2 racks.

There are no restrictions in the placement of the spent fuel assemblies in the rack. Coolant flow rates within the rack are calculated for a maximum powered assembly of 75 kw. The conditions of bulk boiling with no pool cooling is conservatively represented for accident conditions by setting the coolant inlet temperature at the bottom of the racks equal to the saturation temperature at the top of the racks, taking into account the height of the pool surface above the racks. The average coolant temperature rise in the boxes containing the maximum powered assemblies is 20° F. There may be some subcooled boiling within the assemblies, but the enthalpy of the water leaving the top of the rack will be below the enthalpy of saturated liquid. Without any natural circulation between the boxes, there is sufficient natural circulation in the boxes via the space under the racks so that enthalpy of the exit water at the top of the assemblies is below the enthalpy of saturated liquid. Gamma heating in the steel boxes and the water between the boxes is negligible relative to the fuel assembly heating. The channels between the boxes have sufficient flow area and are open at the top and bottom so that natural convection can remove any heat buildup in the system.

2.3.4 RACK MATERIALS AND CONSTRUCTION

The spent fuel racks are manufactured from the following materials:

ASTM-A-240 Sheet and Plate Stock

ASTM-A-276 Bar Stock

ASTM-A-479 Bar Stock

AWS-E-308-15 Weld Wire

AWS-E-308-16 Weld Wire

ASTM-A-351-74B-CF8 Stainless Steel Castings

All welded construction is used in the fabrication of the spent fuel assembly storage rack. This construction assures the structural integrity of the storage modules and provides assurance of smooth, snag-free passage in the storage cavities.

The reactions within the fuel racks are determined for the combined normal operating and maximum seismic loadings. The welds are then sized such that the tensile and shear streeses do not exceed those allowed per Table NF-3292 1-1 of ASME Section III.

The materials, fabrication, installation and examination requirements for the racks are in accordance with NF2000, NF4000 and NF5000 of Section III of the ASME Code for Class 3 component supports. The racks are designed in accordance with the stress allowables in NF-3000 of Section III of the ASME B and PV Code.

2.3.5 QUALITY ASSURANCE

The spent fuel racks are a Seismic Class I, Quality Class 2 structure and are manufactured in accordance with the requirements of CE Specification 00000-WQC 5.2, "Quality Assurance Program" Specification Hi-Cap Fuel Racks.

Regulatory Guide 1.29 states, "The pertinent Quality Assurance requirements of Appendix B to 10 CFR 50 should be applied to all activities affecting the safety related function..."

Combustion Engineering fully complies in all respects with the pertinent requirements of Appendix B 10 CFR 50 for the design, procurement, and fabrication of the spent fuel racks. WQC 5.2 Rev. 0, and the Engineering Specification address the pertinent, applicable portions of 10 CFR 50 Appendix B to cover the fabrication of the spent fuel racks.

The following tabulation indicates the controlling documents which meet the pertinent Appendix B requirements for the design and procurement phase (Windsor column) and the fabrication phase (Fabricator column).

Criterion	Windsor	Fabricator (Supplier)
1	CE POWER SYSTEMS QA Manual	WQC 5.2
II	"	"
III	"	N/A
IV	"	WQC 5.2
V	"	"
VI	"	"
VII	"	"
VIII	N/A	WQC 5.2 & CE Eng. Spec.
IX	CE POWER SYSTEMS QA MANUAL	"
X	"	"
XI	N/A	"
XII	"	WQC 5.2
XIII	CE POWER SYSTEMS QA MANUAL	5.2 & CE Eng. Spec.
XIV	"	"
XI	"	"
XVI	"	"
XVII	"	"
XVIII	"	"

ANSI 45.2.5 "Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants" is not considered applicable to the design, procurement and shop fabrication of the spent fuel racks, since the standard is directed at site-related activities. However, the qualification of welders and welding procedurs for the shop fabrication of these spent fuel racks is in accordance with the ASME Section IX requirements which is acceptable under ANSI 45.2.5.

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TABLE 2.3-1

ALLOWABLE STRESS LIMITS

Linear Type Elastic Stress Analysis based on maximum stress theory was used in the design of fuel storage racks. The allowable stresses are per the requirements of Linear Type Component supports of the ASME Boiler and Pressure Vessel Code Section III, subsection NF.

Applying the rules of the ASME Code Section III, subsection NF for calculating allowable stresses, for linear type supports the following values were obtained:

Normal and Upset (OBE) - Subsection NF, Appen. XVII

F_t = 0.6 Sy = 14250 psi

Fb = 0.6 Sy = 14250 psi

Fv = 0.4 Sy = 9500 psi

Faulted Condition (SSE) - Appen. XVII, Appen. F

F_t = (2) (0.6) Sy = 28500 psi

Fb = (2) (0.6) Sy = 28500 psi

Fv = (2) (0.4) Sy = 19000 psi

Yield strength values Sy for high alloy steels for Class 3 components are listed in Appendix I, Table I - 2.2 of the ASME Code.

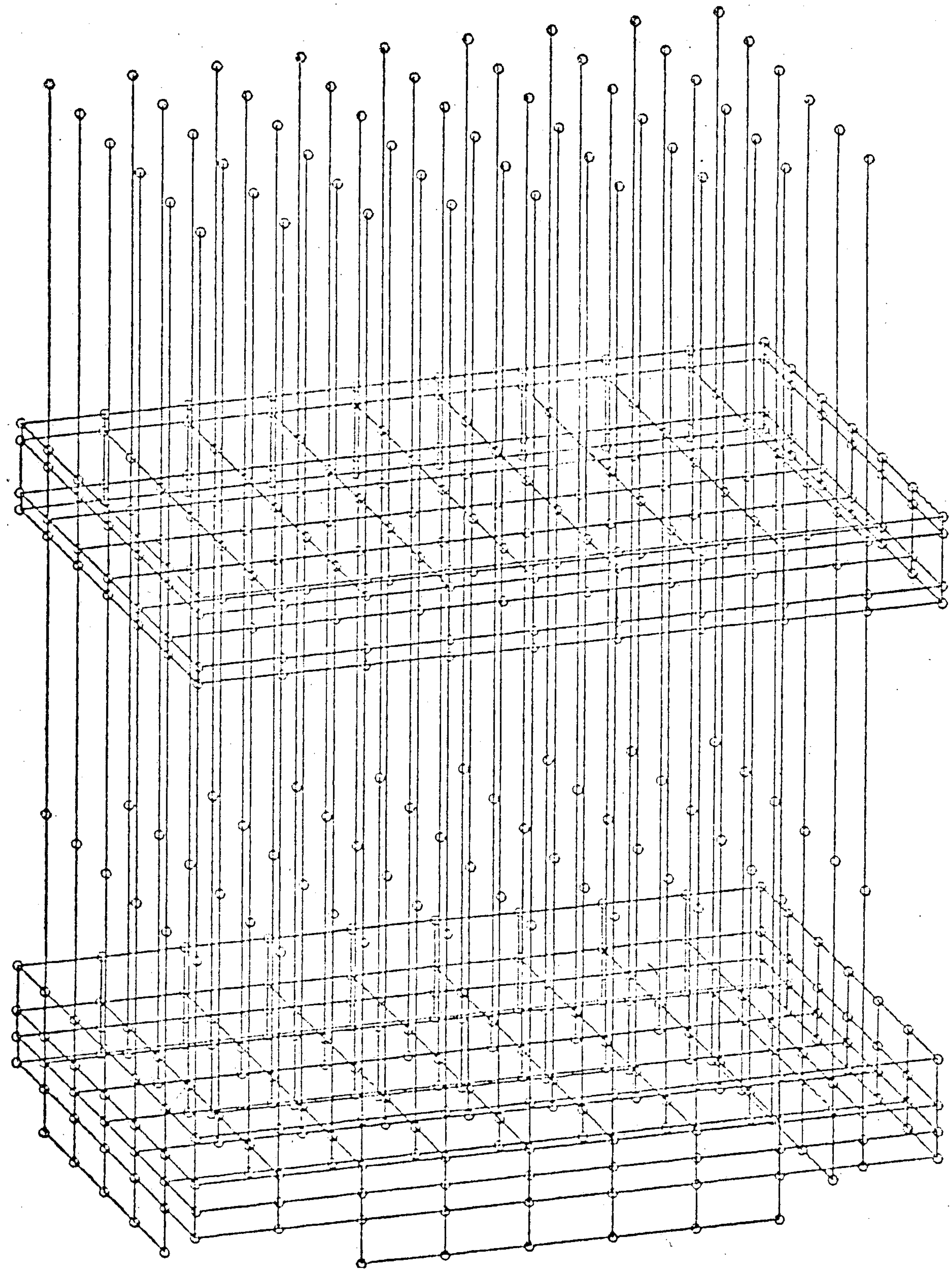
Allowable stress limits for welds in linear type supports is stipulated in subsection NF, Article NF-3292.1, Table NF-3292. 1-1.

TABLE 2.3-2

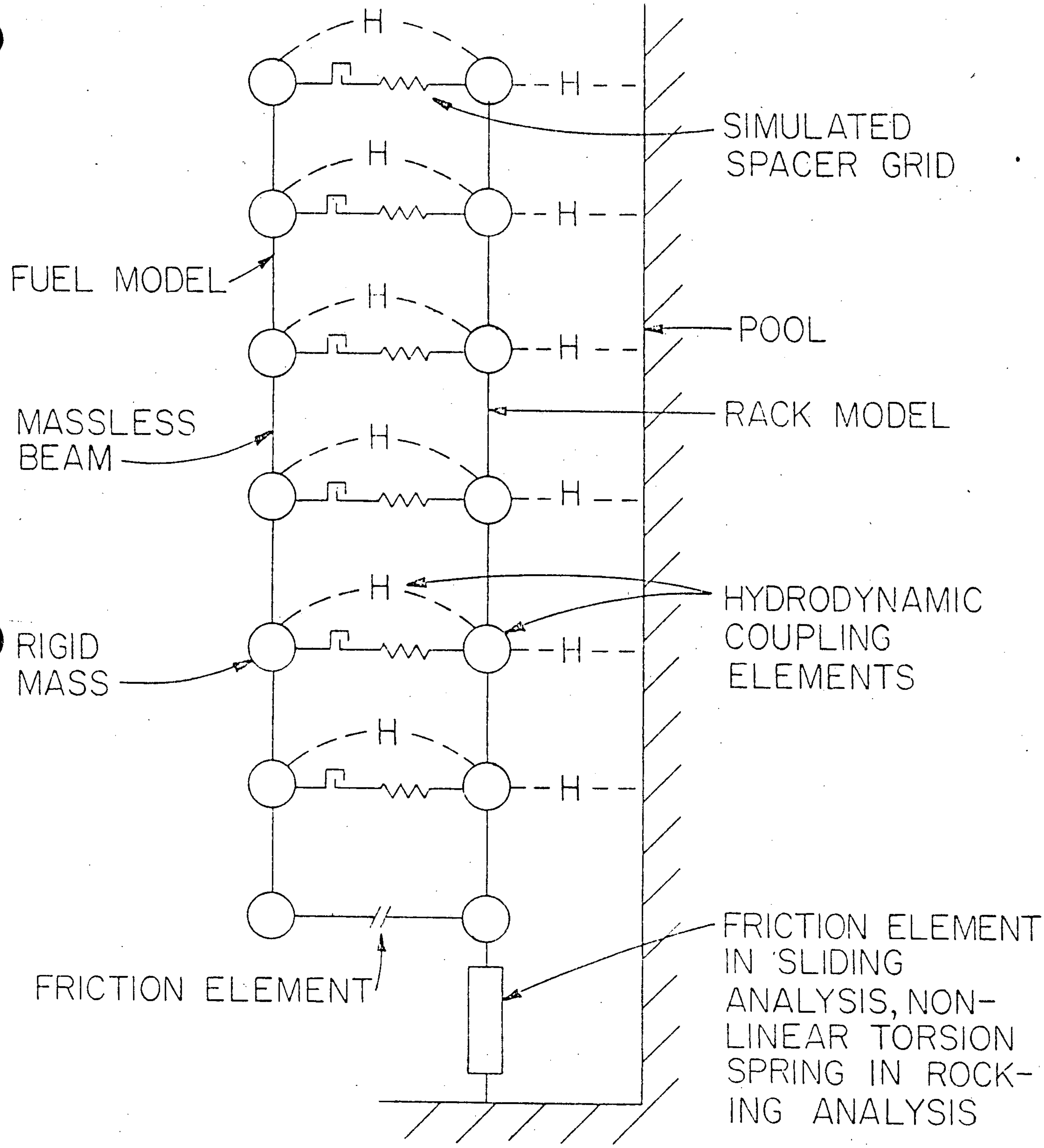
RESULTS OF ANALYSIS OF CRITICAL UO_2 SYSTEMS

LATTICE		w/o U-235	PITCH (in.)	H_2/UO_2	BORON (ppm)	K_{eff}	Ref
B&W-1273	1	4.020	0.595	1.137	0	0.998	9
	2	4.020	0.595	1.137	3390	1.0018	9
	3	4.020	0.571	0.056	0	0.9963	9
	4	2.2459	0.595	1.371	0	1.0009	9
	5	2.459	0.595	1.371	1675	1.0016	9
B&W-3647	6	2.459	0.644	1.846	0	1.0004	10
	7	2.459	0.644	1.846	846	1.0014	10
	8	2.459	0.644	1.846	1536	0.9997	10
Yankee	9	2.700	0.405	1.048	0	0.9995	11
	10	2.700	0.435	1.405	0	0.9979	11
	11	2.700	0.470	1.853	0	0.9990	11
	12	2.700	0.493	2.166	0	1.0004	12
Winfrith	13 (20C)	3.003	0.520	1.001	0	0.9987	13
(80C)	14	3.003	0.520	1.001	0	0.9977	13
	15	3.003	0.735	3.164	0	1.0009	13
	16	3.003	0.492	0.779	0	0.9992	13
Bettis	17	1.311	0.6133π	1.429	0	0.9963	14
	18	1.311	0.6133π	1.429	0	0.9963	14
	19	1.311	0.6133π	1.429	0	0.9970	14
	20	1.311	0.6504π	1.781	0	0.0962	14
	21	1.311	0.6504π	1.781	0	0.9975	14
	22	1.311	0.7110π	2.401	0	0.9968	14
	23	1.311	0.7110π	2.401	0	0.9957	14

π Triangular Pitch



SPACE FRAME MODEL OF FUEL RACK
OCONEE NUCLEAR STATION
Figure 2.3-1



FUEL RACK NON-LINEAR SHOCK MODEL
OCONEE NUCLEAR STATION
Figure 2.3-2

3.0 SPENT FUEL POOL/RACK INTERFACE

3.1 STRUCTURAL

The spent fuel pool and its cooling system are described in the Oconee Nuclear Station Final Safety Analysis Report Section 9.4. Figure 3.1-1 is a reproduction of FSAR Figure 9-11. It shows the general arrangement of the Units 1 and 2 pool and the associated fuel handling equipment. This arrangement will not be changed as a result of this modification.

The spent fuel pool is constructed of concrete with a compressive strength of 3000 psi at 28 days and reinforcing steel with a minimum strength of 60,000 psi. The spent fuel pool liner is constructed of ½" stainless steel clad plate. The fuel pool concrete reinforcing steel, liner plate and welds connecting the liner plate to the fuel pool floor concrete embedments will be analyzed based on consideration of the new racks and additional fuel. Design criteria including loading combinations and allowable stresses shall be in compliance with Oconee FSAR Appendix 5A for Class I structures. The determination of Ta (abnormal thermal load condition) to be used in combination with E' is based on the failure of one pump or cooler during normal operating conditions.

3.2 THERMAL

3.2.1 DESIGN BASES

The Spent Fuel Pool Cooling System are designed to keep the pool water temperature below 150°F for normal refueling operations and full core discharge situations. Under normal refueling conditions the fuel is discharged over a four day period after at least three days cooling inside the reactor vessel. The full core discharge is expected to take four days also with three days cooling in the reactor vessel prior to moving any fuel. The heat released from the fuel stored in the pool is determined in accordance with Branch Technical Position APCSB 9-2 "Residual Decay Energy for Light Water Reactors for Long Term Cooling." Table 3.2-1 shows the expected loading in the pool. In addition to the normal discharges the pool will retain the capability to accept a full core discharge. The heat load from a full core discharge in addition to normal heat loading is shown in Table 3.2-2. In the event mixed oxide fuel becomes available the heat load in the pool will be slightly higher. The increase is apparent only in fuel which has decayed for a relatively long period of time, and contributes little additional heat load to the pool.

3.2.2 SYSTEM DESCRIPTION

The Spent Fuel Cooling System is described in Section 9.4.2. of the Oconee FSAR The system is designed to provide sufficient cooling for a full core offload of fuel, a recently discharged 1/3 core and all the remaining locations full of decayed fuel as described in FSAR Section 9.4.2. This system will continue to be sufficient until additional capacity can be installed. It is anticipated that such installation will take place in the first quarter of 1980. A modification to add an additional cooler and pump which will take suction from existing coolant piping will be completed prior to the number of fuel assemblies in the pool exceeding the original design capacity. Heat exchanger cooling water will be drawn from the recirculating water (RCW) system. The added cooler and pump are described in Table 3.2-3.

3.2.3 DESIGN EVALUATION

During normal operation the Spent Fuel Pool Cooling System serves two main functions. The first is to maintain the pool water at temperatures below 150°F. The second function is to provide purification of the spent fuel pool coolant for clarity during fuel handling operations. Under normal conditions, the pool temperature is maintained under 125°F as stated in Oconee FSAR Section 9.4.2.1 by recirculating spent fuel cooling water from the spent fuel pool through the pumps and coolers and back into the pool. The pumps and coolers are arranged in parallel. The purification function is performed as described in the Oconee FSAR Section 9.4.2.1.1.

The heat loads shown in Tables 3.2-1 and 3.2-2 represent the largest heat loads expected in the spent fuel pool. The normal case assumes that Unit 1 and 2 are refueled consecutively and the pool is filled with previous discharges except those spaces reserved for full core discharge. The full core discharge case assumes consecutive refueling followed by a full core discharge after a short period of operation. In this case all the spaces in the pool are filled. The calculations conservatively assume that 18 month cycles are used on both units.

Pool temperature is maintained below 150°F by operation of any two pump-cooler configurations for the normal heat load and by operation of all three pumps and coolers for the maximum heat load. Upon failure of one pump or cooler for either of these conditions sufficient cooling capacity remains to maintain bulk pool temperature below 205°F. An analysis of pool response to loss of all forced cooling is presented in Section 6.4 of this document.

3.3 WATER QUALITY

Operating experience has shown that concentrations of radionuclides are greatest during periods of fuel movement in the pool (i.e., refueling) and are not directly related to the number of assemblies stored in the pool. Therefore, the increased load on the Spent Fuel Pool Purification System will be small and the existing system will adequately maintain water chemistry, clarity, and activity within acceptable levels.

Table 3.2-1

NORMAL HEAT LOADS FOR OCONEE 1 AND 2 SPENT FUEL POOL

Number of Assemblies	Irradiation (EFPD)	Decay Time	Power Fraction (10^{-3})	Heat Output (10^6 BTU/hr)
72	1263	7 Days	3.0	10.7
72	1263	35 Days	1.53	5.4
144	1263	1.5 Years	.22	1.6
144	1263	3.0 Years	.12	.9
144	1263	4.5 Years	.10	.7
69	1263	6.0 Years	.09	.3
645				19.6

Table 3.2-2

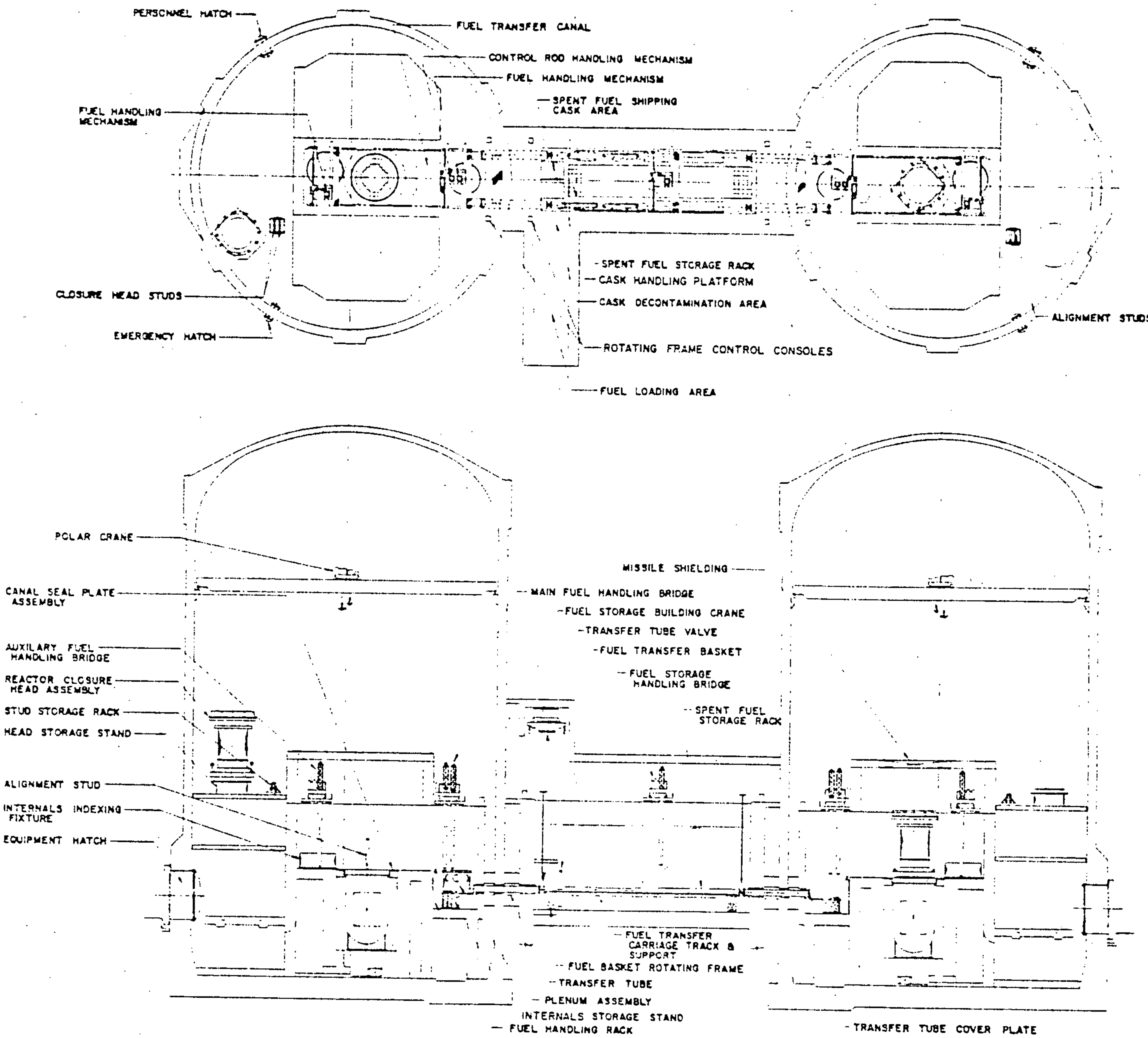
MAXIMUM HEAT LOADS FOR OCONEE 1 AND 2 SPENT FUEL POOL

Number of Assemblies	Irradiation (EFPD)	Decay Time	Power Fraction (10^{-3})	Heat Output (10^6 BTU/hr)
72	11	7 Days	1.16	4.1
72	432	7 Days	2.83	10.1
33	853	7 Days	2.96	4.8
72	1263	35 Days	1.53	5.4
72	1263	60 Days	1.13	4.1
144	1263	1.5 Years	.22	1.6
144	1263	3.0 Years	.12	.9
141	1263	4.5 Years	.1	.7
750				31.7

Table 3.2-3

COMPONENTS ADDED TO OCONEE UNITS 1 AND 2 SPENT FUEL POOL COOLING SYSTEM

Spent Fuel Cooler	
Type	Tube & Shell
Material Tube/Shell	CS / SS
Capacity, BTU/hr/cooler	7.75×10^6
Cooling Water Flow lbm/hr/cooler	5×10^5
Code	ASTME VIII/III-C
Spent Fuel Pump	
Type	Horizontal Centrifugal
Material	SS
Flow, GPM	1000
Head, ft. H_2O	100
Motor, hp	40



FUEL HANDLING SYSTEM
OCONEE NUCLEAR STATION
Figure 3.1-1

4.0 RACK INSTALLATION

The installation plan is based on the following objectives:

(a) Maintaining installation exposure levels As Low As Reasonably Achievable (ALARA).

(b) Avoiding transport of rack modules over stored fuel.

(c) Achieving acceptable tolerances on module verticality, levelness and positioning.

The installation plan described below is considered to be the most effective means to address these concerns.

4.1 REMOVAL OF EXISTING RACKS AND INSTALLATION OF NEW RACKS

The existing racks are of the "open type" constructed of stainless steel pipe and angles. The configuration of these existing racks is shown in Figure 4.2-1.

Underwater divers will be used to cut the pipe supports of the existing racks approximately 1" above the embedded floor plate to which they are welded. Connecting angles will then be cut and rack sections no larger than 5' x 7' x 14' will be removed from the pool and taken to a controlled work area in the fuel loading area of the Auxiliary Building. At this location, the racks will be further sectioned into pieces adequate for packaging and disposal, as necessary.

The total volume of material to be disposed will occupy approximately 800 cu. ft. and will weigh approximately 23,000 pounds.

All underwater cutting operations will be performed while using an underwater vacuuming system with shielded filters. Each portion of the existing rack structure will be rinsed off as it is removed from the pool.

Figures 4.1-2 through 4.1-11 depict the step by step procedures which is anticipated to be utilized in removal of existing racks and in the installation of the new modular racks. As shown in these diagrams, approximately 140 spent fuel assemblies will be moved to various storage locations in the pool during the reracking operation. The distance between divers performing work and stored fuel (i.e., open area) will be maximized at all times. The step-by-step process anticipated is described as follows:

STEP	ACTIVITY
1	Store all 140 assemblies in existing racks at the south end of the pool.
2	Remove approximately 1/3 of the existing racks at the north end of the pool.
3	Install 2 new modules in the north end.

STEP	ACTIVITY
4	Transfer 44 assemblies to the new storage rack modules.
5	Remove an additional portion of the existing racks.
6	Transfer 24 assemblies back to the existing storage locations in the south end.
7	Install an additional new module.
8	Transfer 72 assemblies to the new storage modules.
9	Remove an additional portion of the existing racks.
10	Transfer 22 assemblies back to the south end of the pool.
11	Install an additional new module.
12	Transfer all assemblies to the new storage modules installed in the north end.
13	Remove the remaining existing racks in the pool.
14	Transfer assemblies within the new modules.
15	Install 4 new modules in the south end.
16	Transfer 48 assemblies to the new modules in the south end.
17	Install 2 additional modules.
18	Transfer the remaining 92 assemblies to the south end.
19	Install remaining additional new modules.

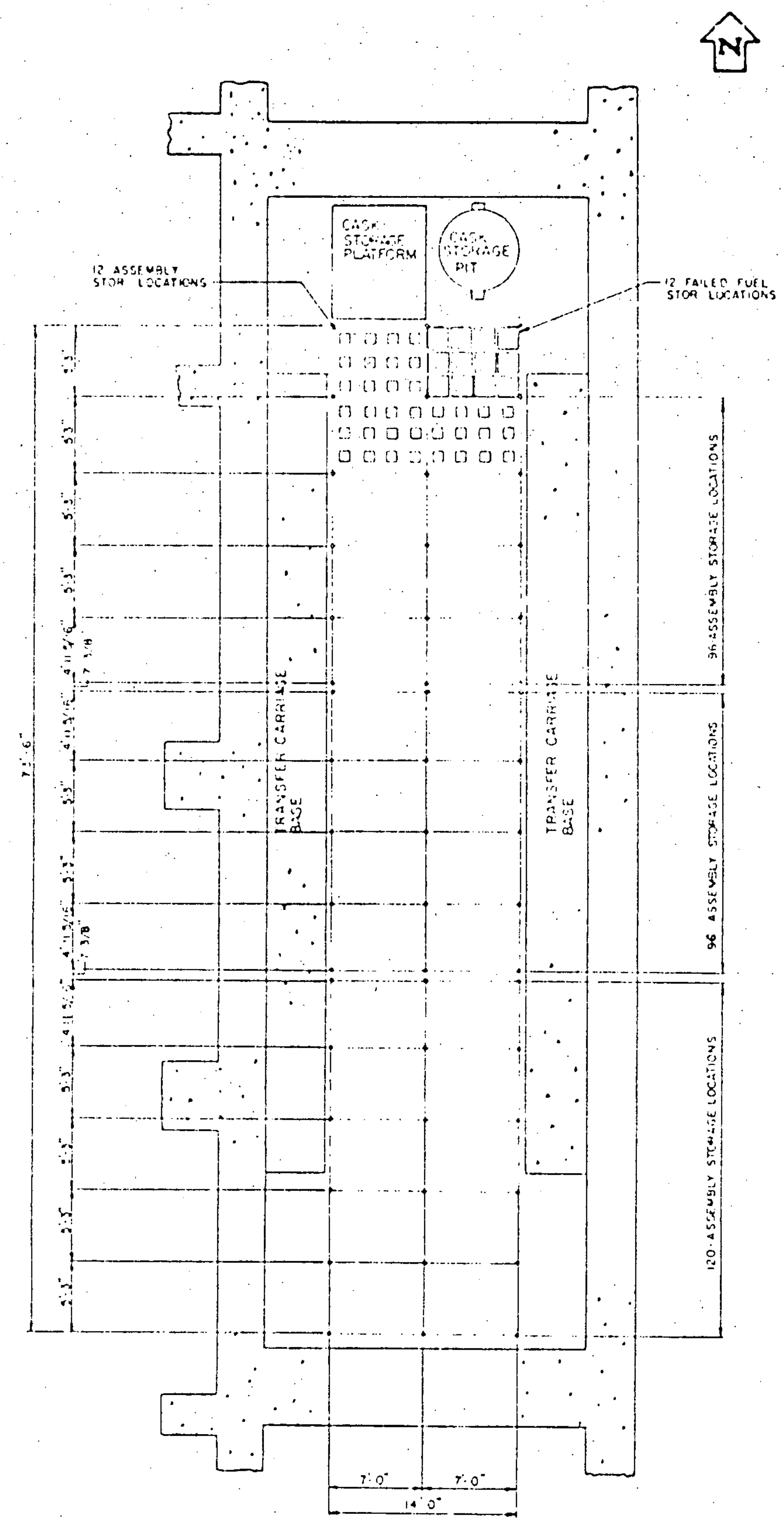
The final configuration of the 14 new modules is shown in Figure 4.1-11. All new modules will be brought preassembled into the fuel handling area by truck and lifted to the cask platform by the 100 ton cask handling crane. The modules will then be rerigged to a temporary construction crane. The construction crane will move all modules underwater along the path shown in Step 14 of Figure 4.2-8. The module will be lowered to an elevation below the top of any stored fuel, but above the fuel transfer carriage for underwater travel to its final location. At no time will construction crane loads be moved over fuel stored in the pool.

The bearing pads supporting each module will be fabricated based on field survey measurements to insure proper levelness of each module. Each rack module will be checked to insure that levelness and position are within design tolerances prior to the placement of the next module.

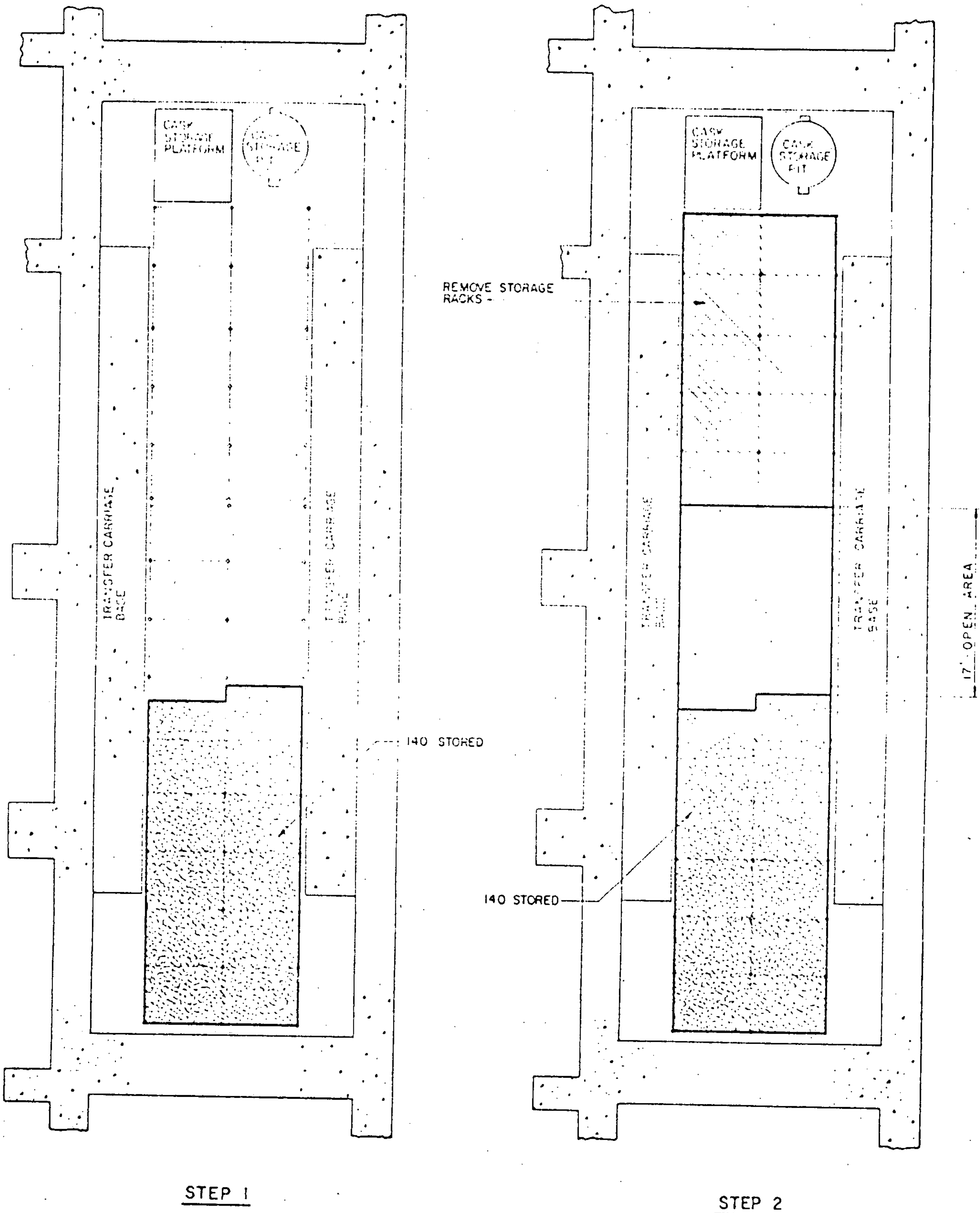
The base of existing racks consisting of bent plates welded to the bottom liner will remain in place unless upexpected circumstances arise during re-racking that requires their removal. The bearing pad thickness for the new racks will position the racks above the existing bent plates.

4.2 QUALITY ASSURANCE

The quality assurance aspects of the removal of the old racks and the installation of the new racks will be carried out in such a manner as to meet the applicable requirements of the Duke Power Company Quality Assurance Program as described in Topical Report DUKE-1A.

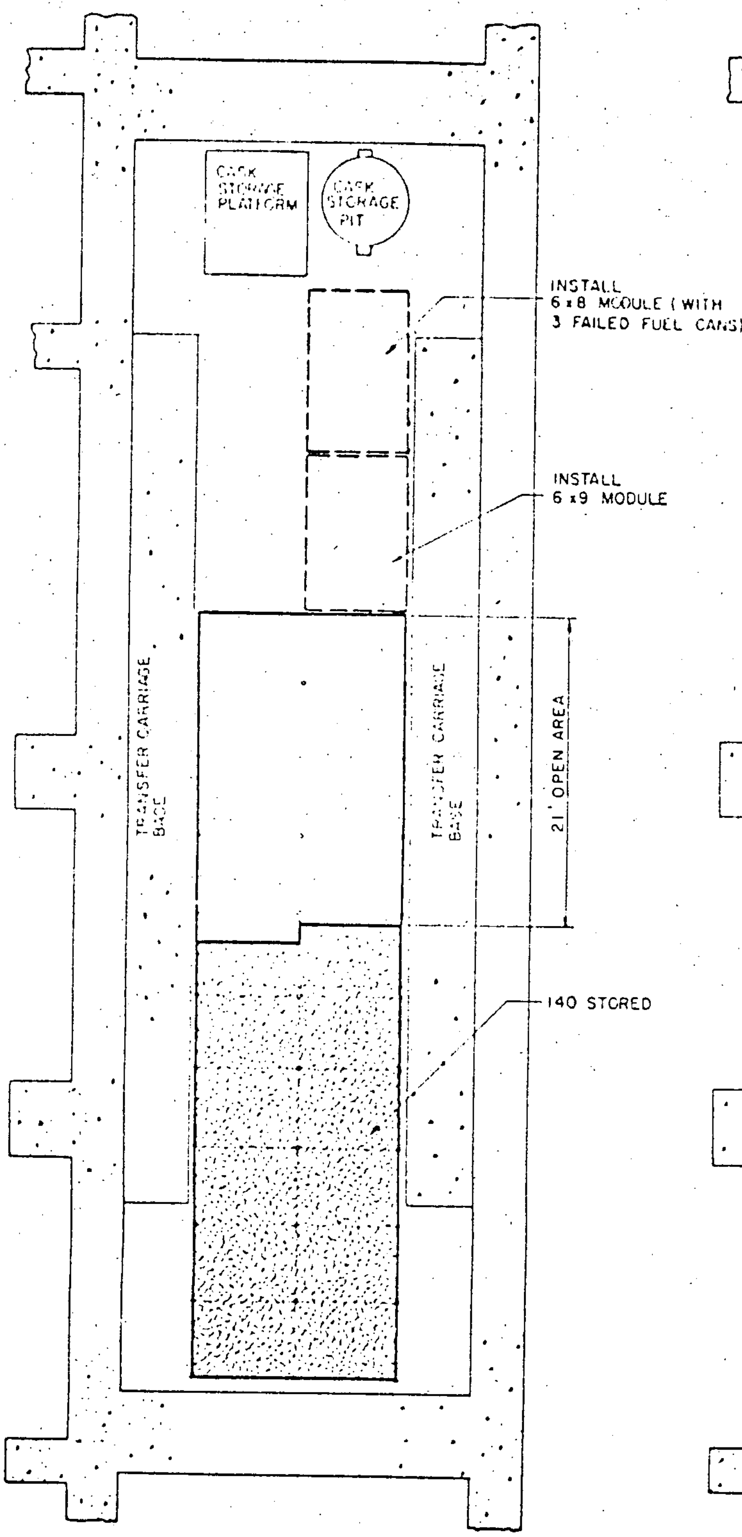


UNIT 1 AND 2 SPENT FUEL POOL
EXISTING CONFIGURATION
OCONEE NUCLEAR STATION
Figure 4.1-1

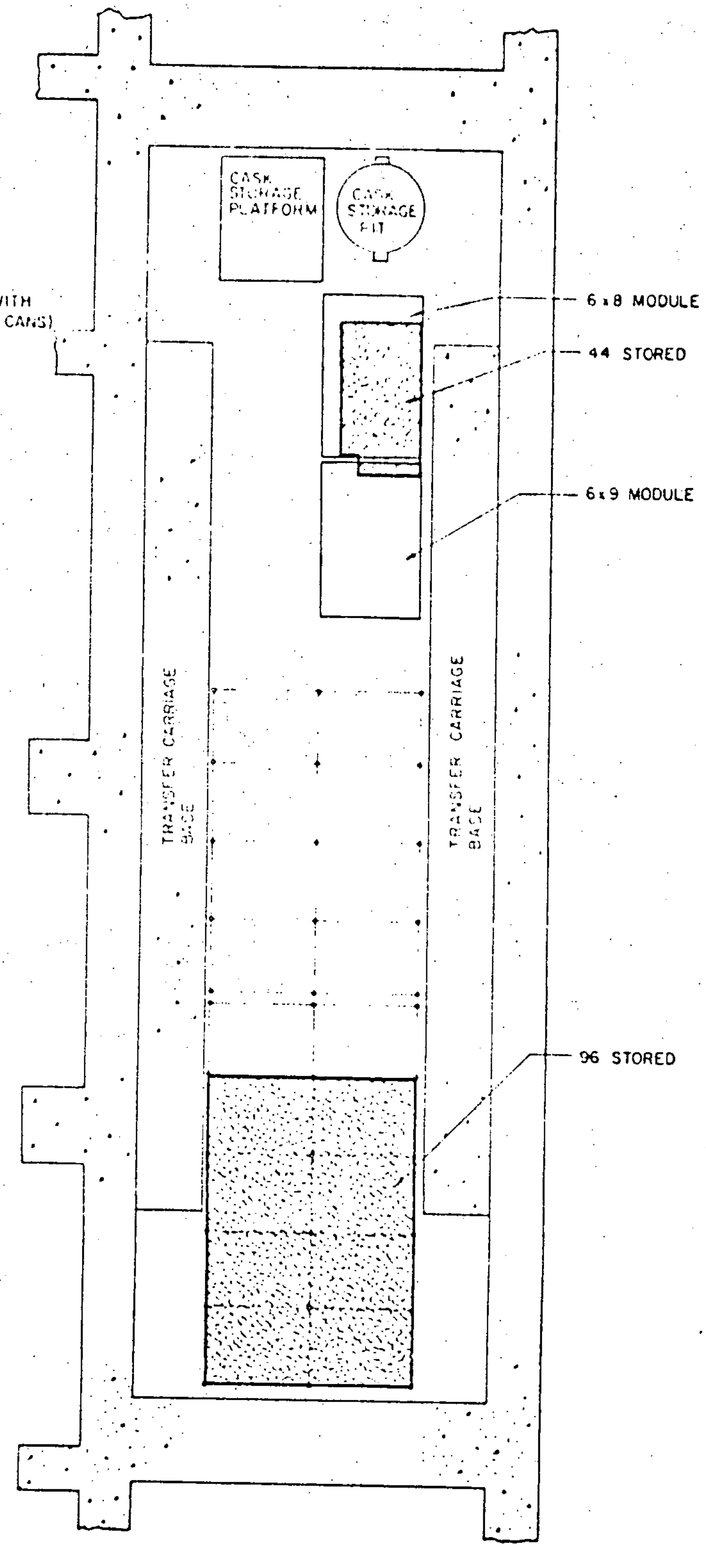


UNITS 1 AND 2 SPENT FUEL POOL
RERACKING PROCESS
OCONEE NUCLEAR STATION

Figure 4.1-2



STEP 3

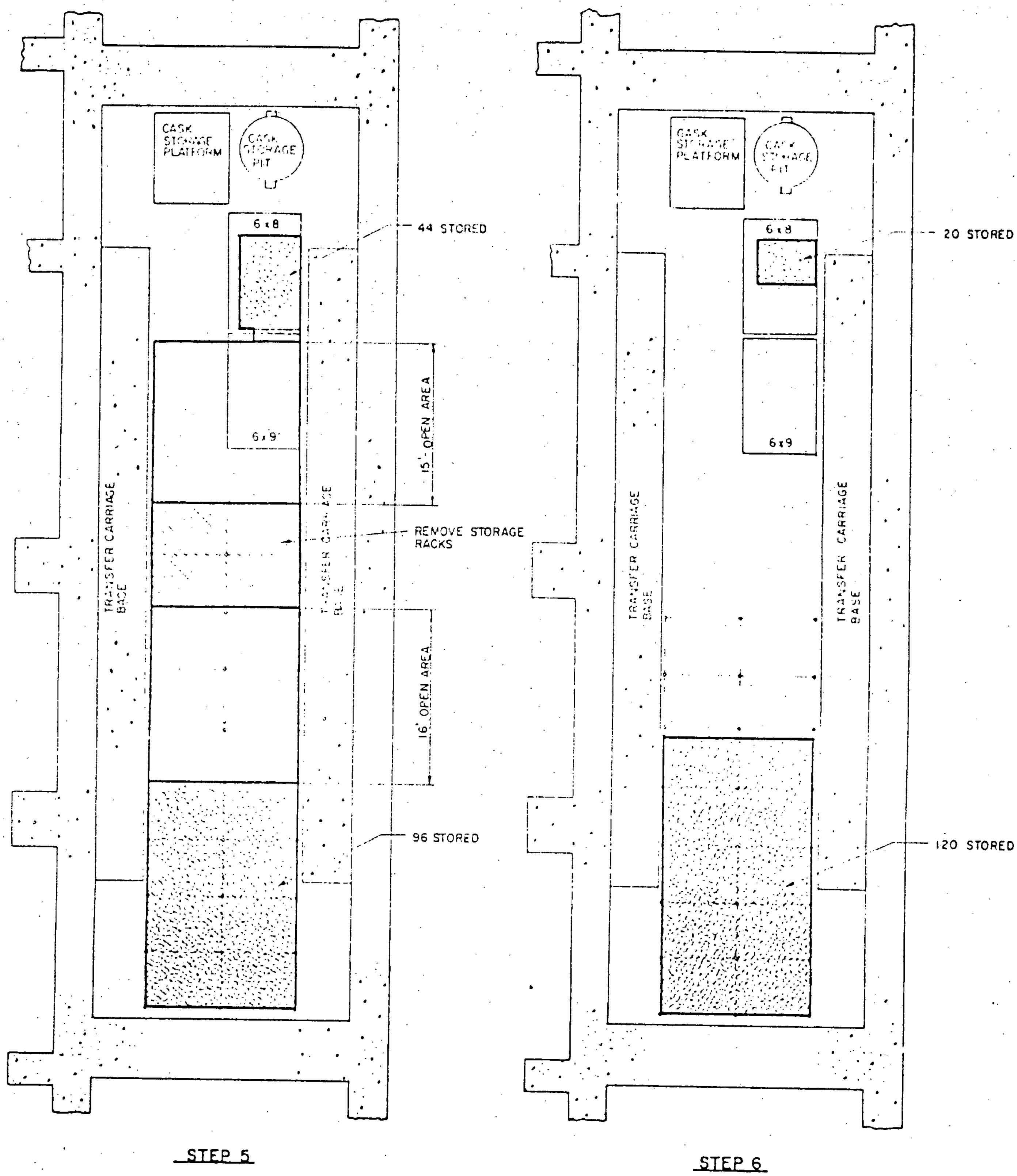


STEP 4

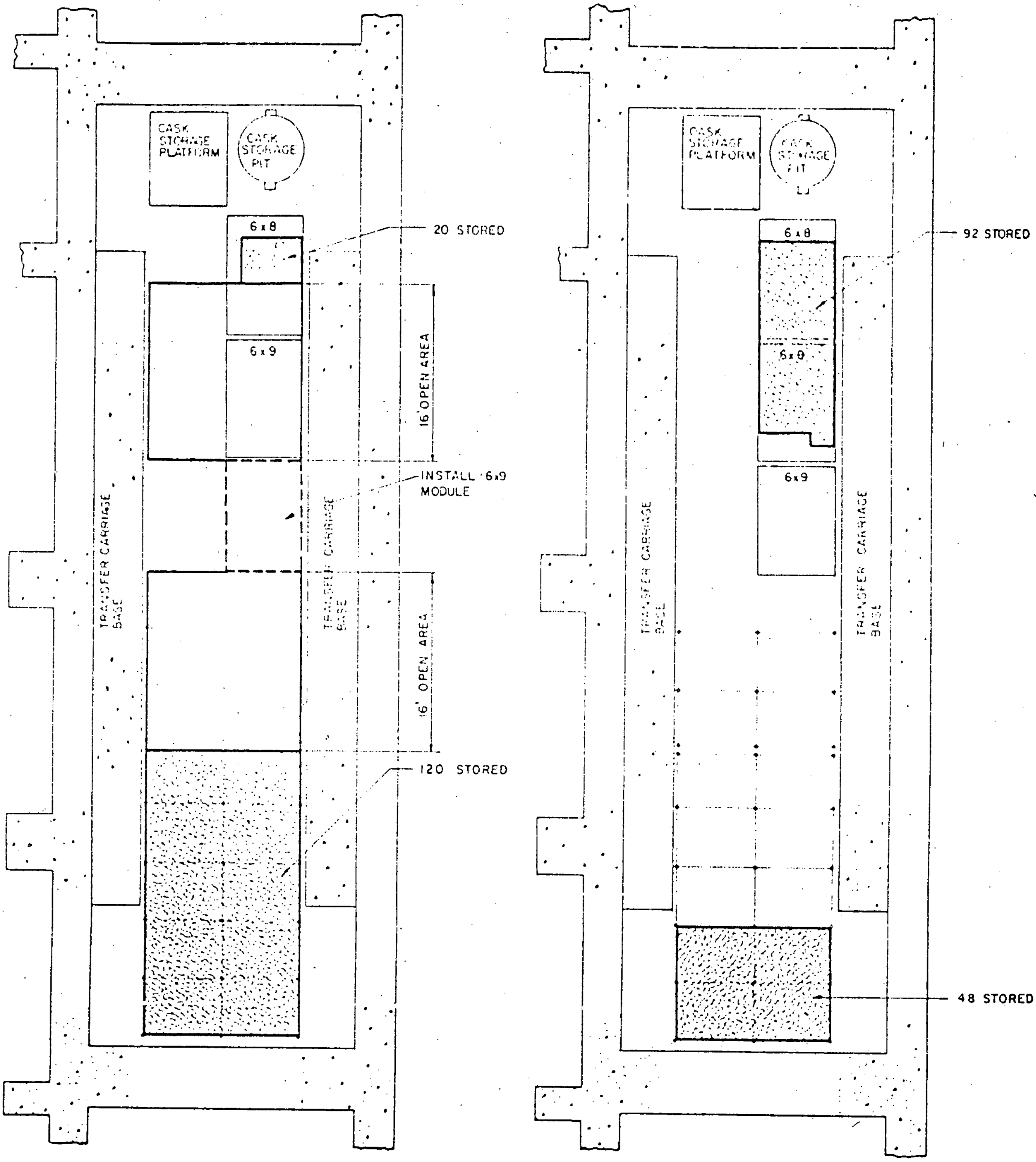
UNITS 1 AND 2 SPENT FUEL POOL RERACKING PROCESS

OCONEE NUCLEAR STATION

Figure 4.1-3



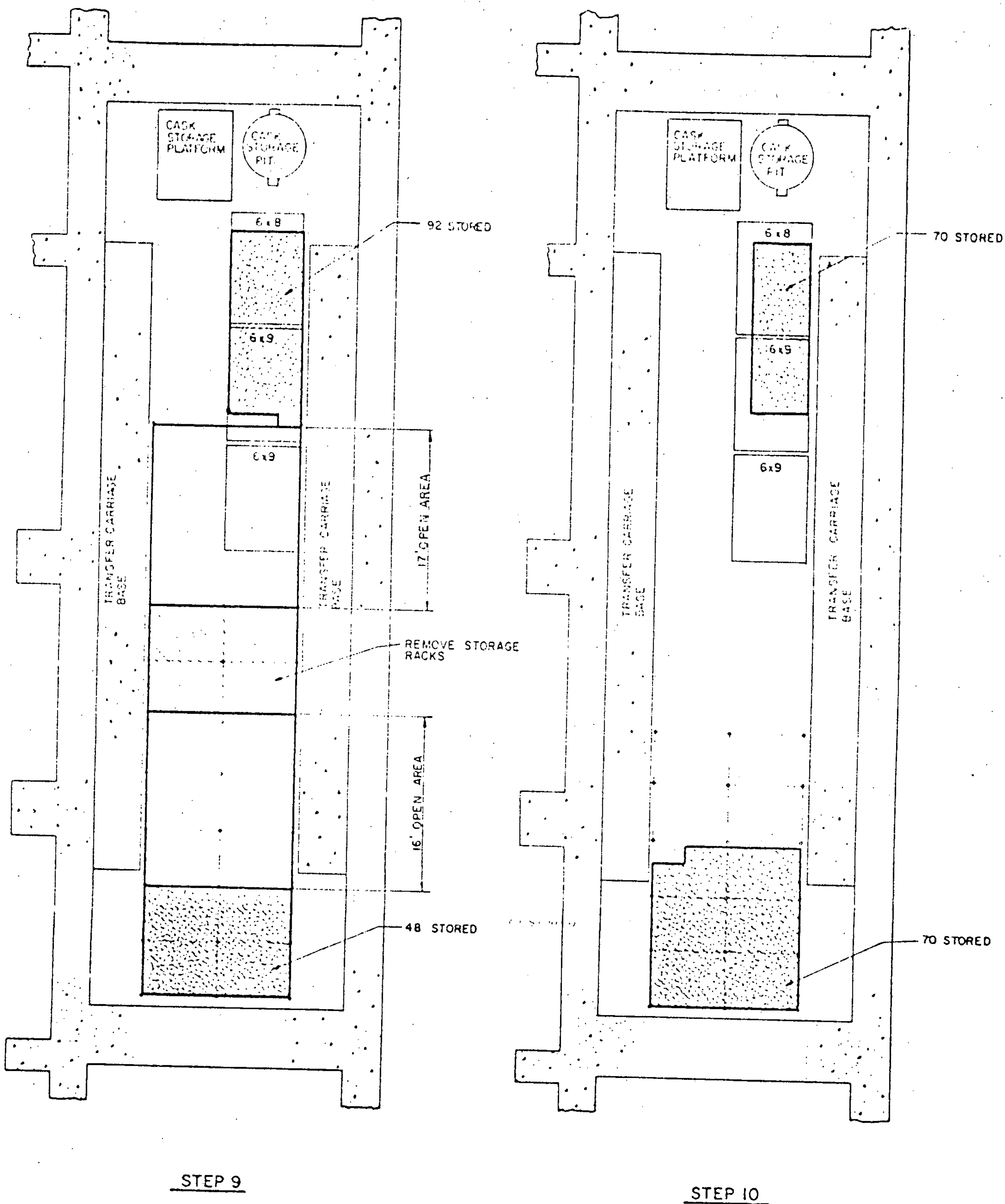
UNITS 1 AND 2 SPENT FUEL POOL RERACKING PROCESS
OCONEE NUCLEAR STATION
Figure 4.1-4



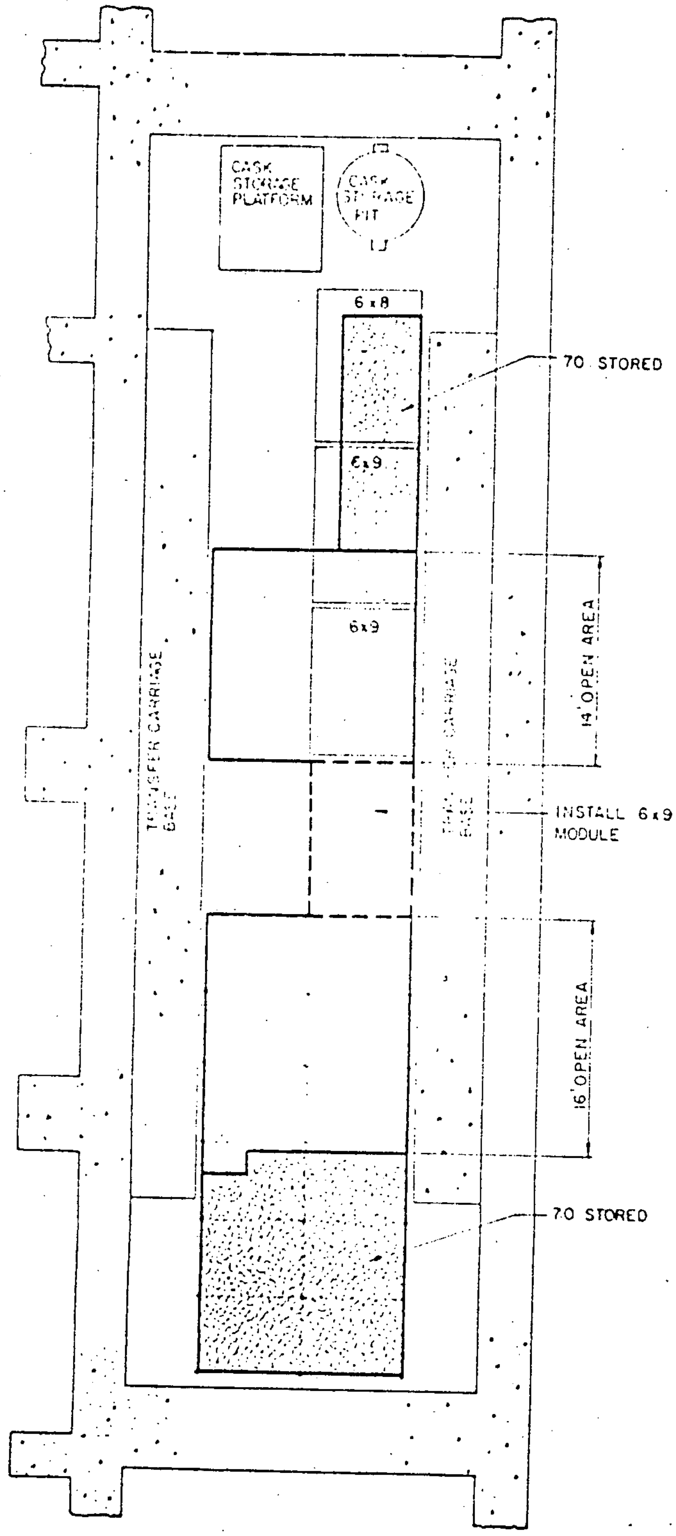
STEP 7

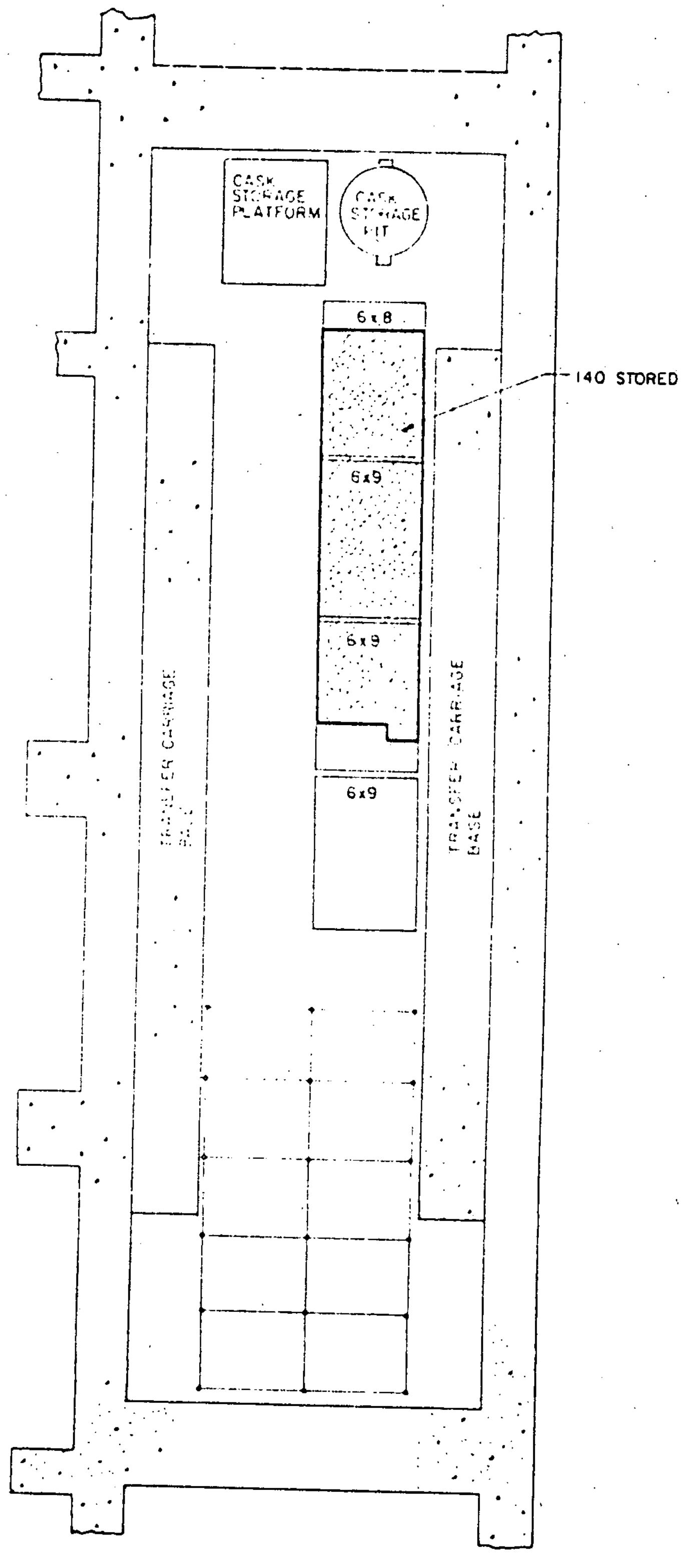
STEP 8

UNITS 1 AND 2 SPENT FUEL POOL
RERACKING PROCESS
OCONEE NUCLEAR STATION
Figure 4.1-5

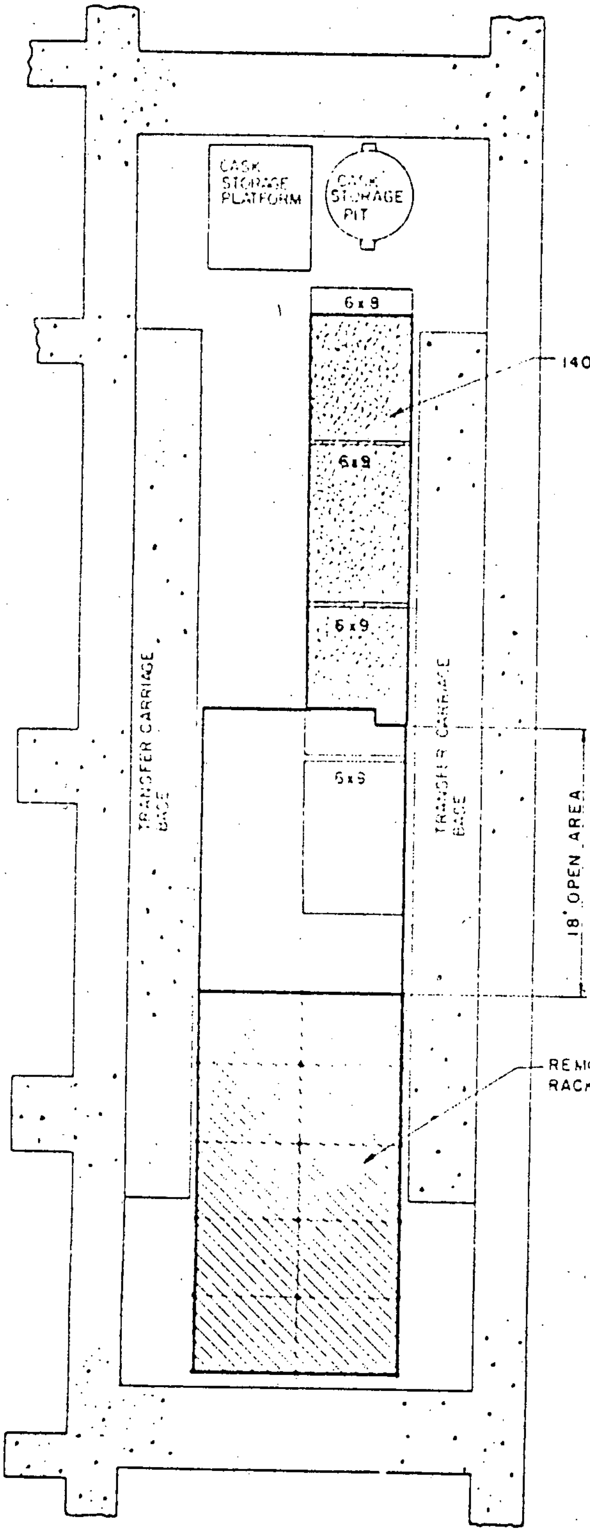


UNITS 1 AND 2 SPENT FUEL POOL
RERACKING PROCESS
OCONEE NUCLEAR STATION
Figure 4.1-6

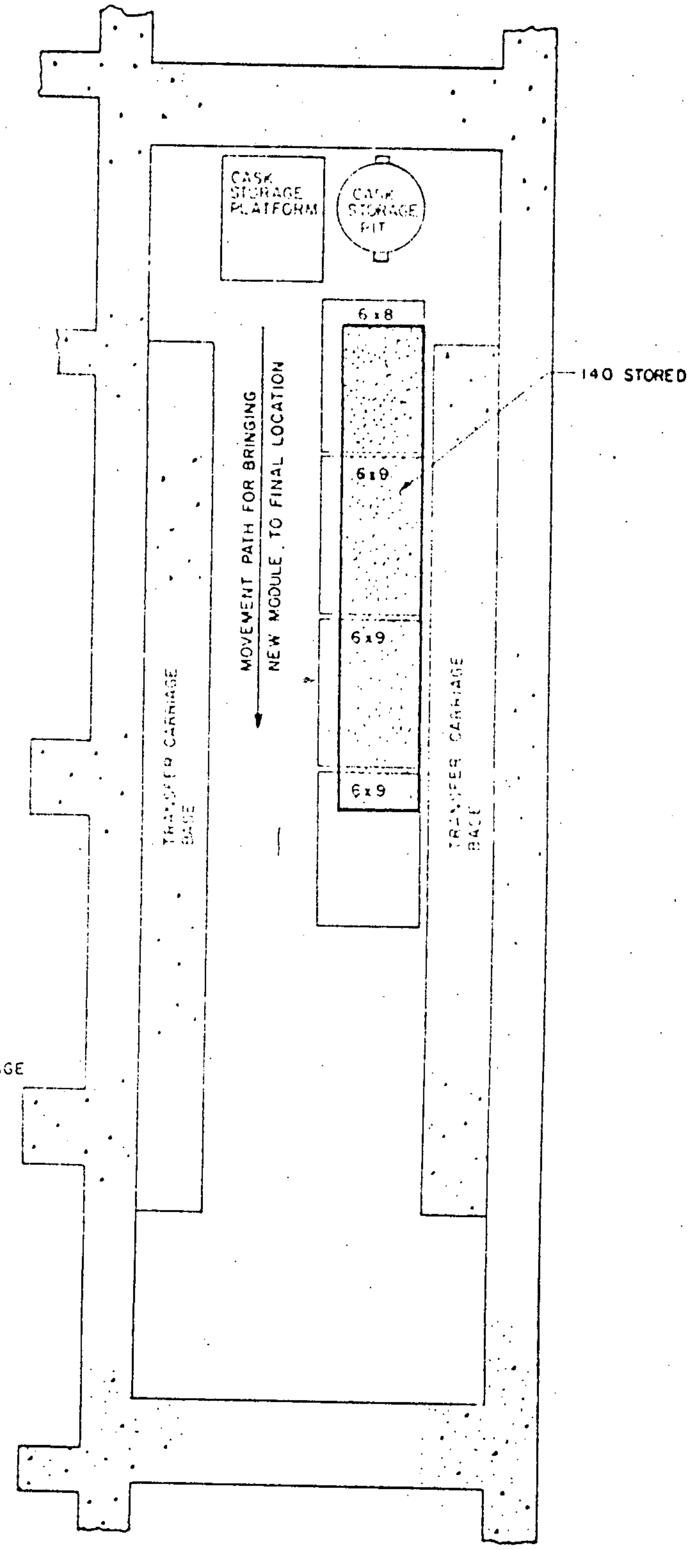




UNITS 1 AND 2 SPENT FUEL POOL
RERACKING PROCESS
OCONEE NUCLEAR STATION
Figure 4.1-7

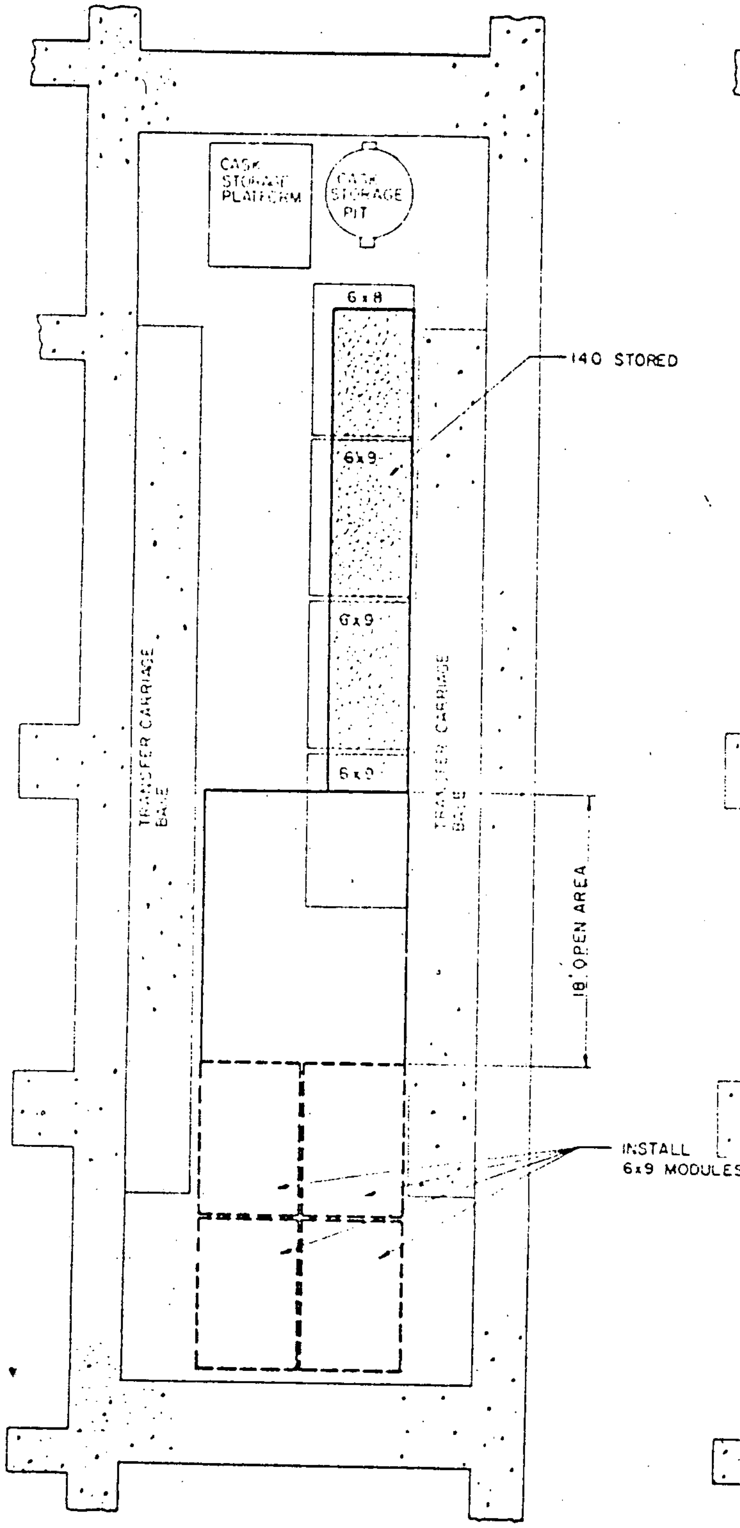


STEP 13

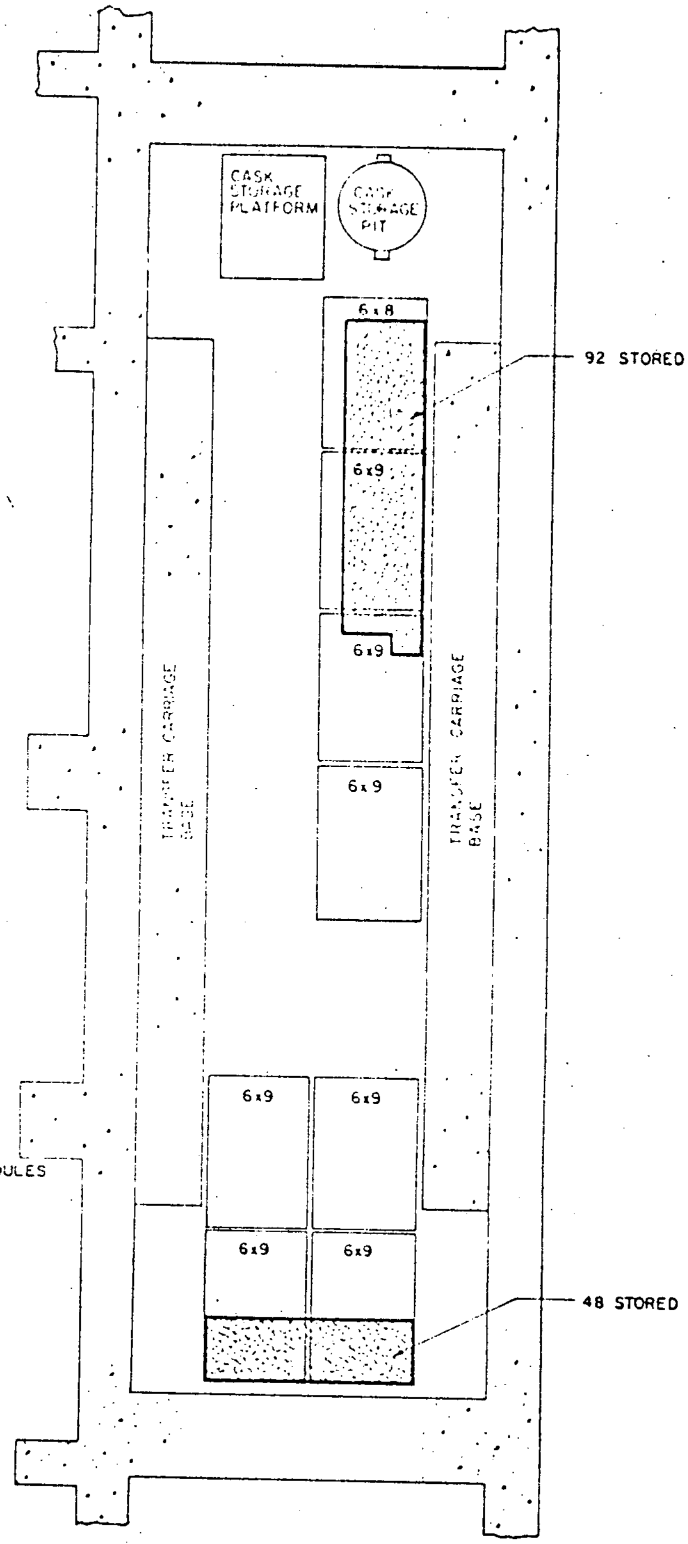


STEP 14

UNITS 1 AND 2 SPENT FUEL POOL
RERACKING PROCESS
OCONEE NUCLEAR STATION
Figure 4.1-8

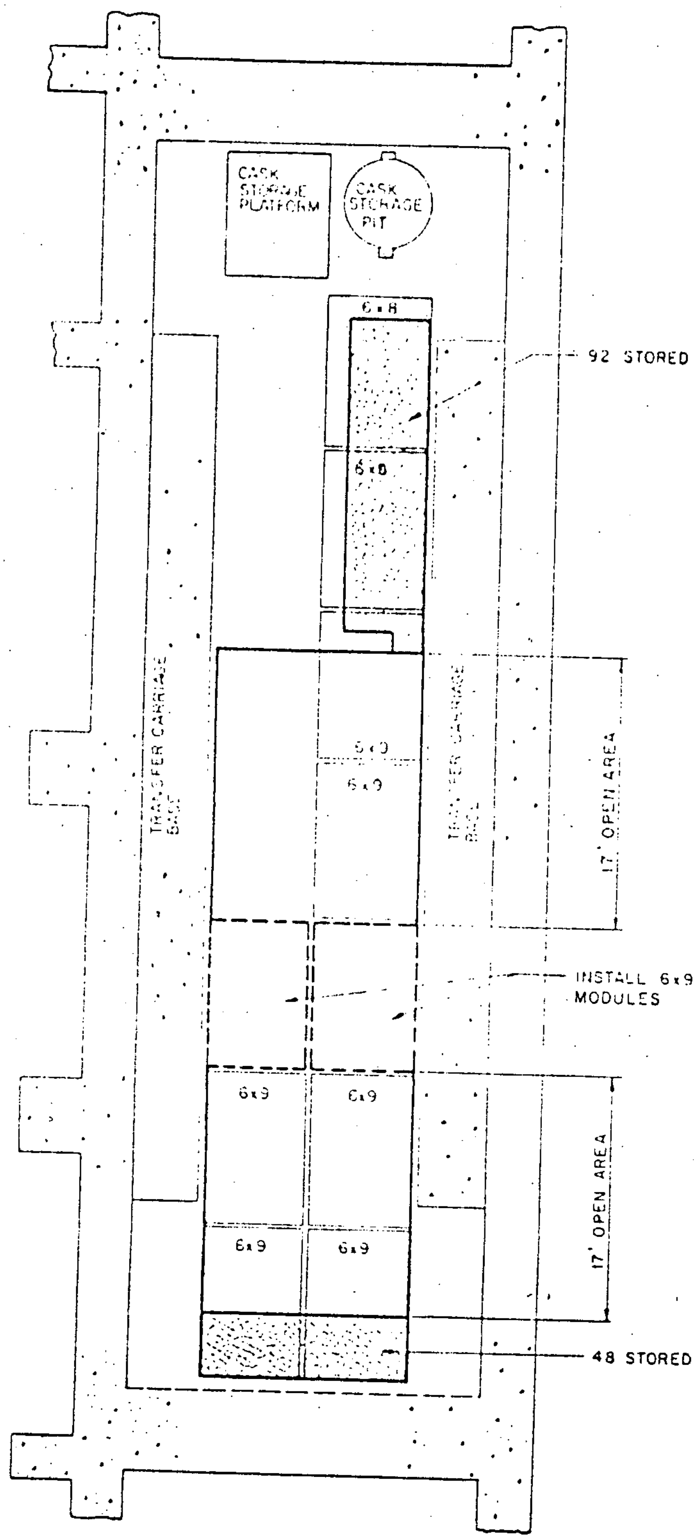


STEP 15

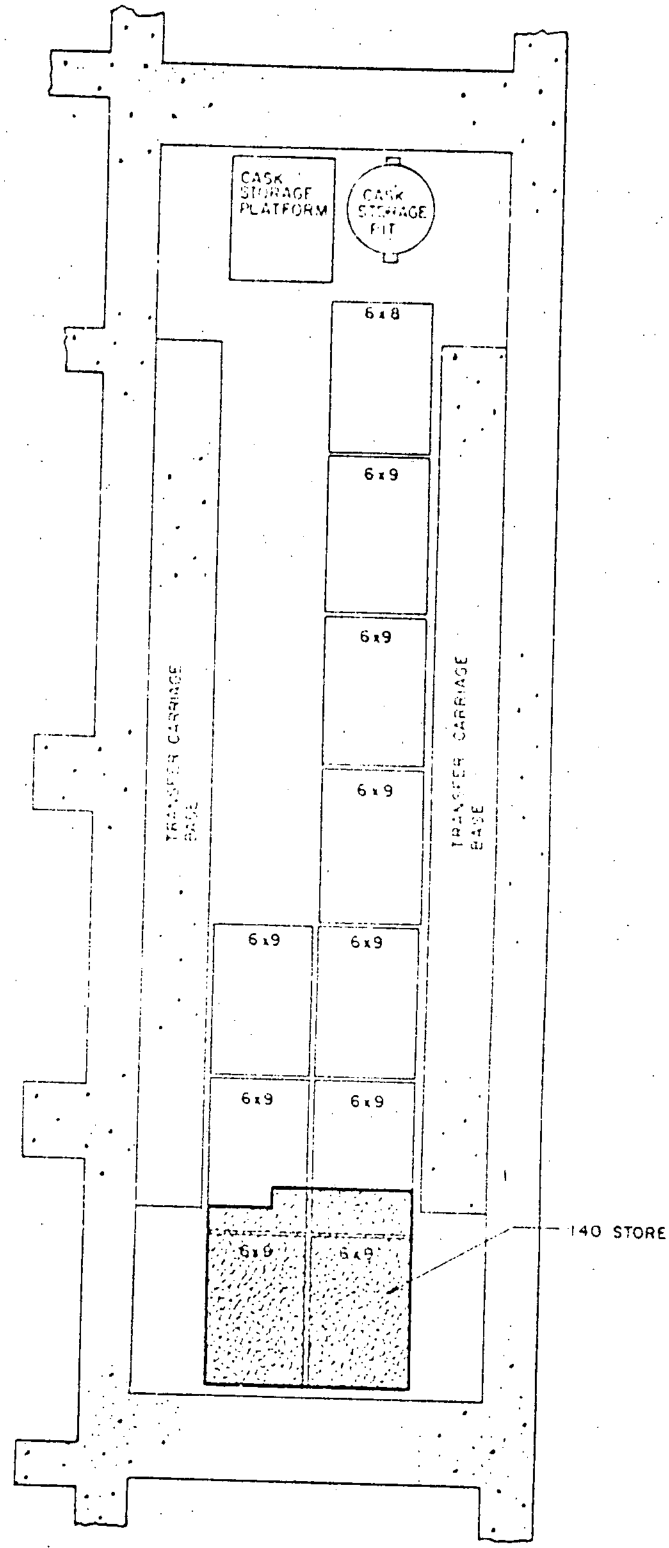


STEP 16

UNITS 1 AND 2 SPENT FUEL POOL
RERACKING PROCESS
OCONEE NUCLEAR STATION
Figure 4.1-9

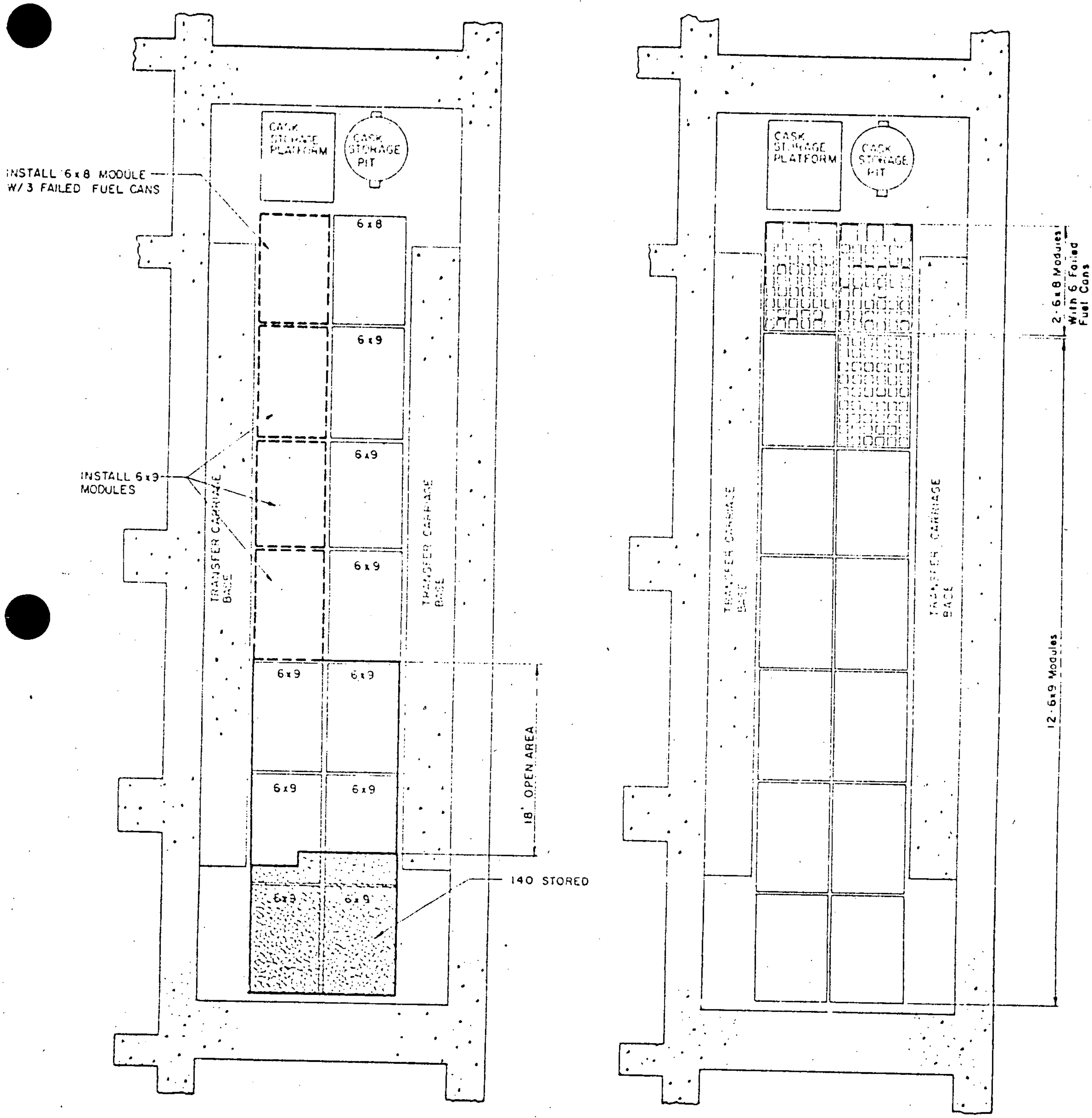


STEP 17



STEP 18

UNITS 1 AND 2 SPENT FUEL POOL REREACKING PROCESS
OCONEE NUCLEAR STATION
Figure 4.1-10



STEP 19

FINAL CONFIGURATION

UNITS 1 AND 2 SPENT FUEL POOL RERACKING PROCESS
OCONEE NUCLEAR STATION
Figure 4.1-11

5.0 RADIATION PROTECTION

5.1 GENERAL DESCRIPTION OF RADIOLOGICAL ASPECTS OF PROJECT

The radiation protection aspects of the spent fuel pool modification are the responsibility of the Station Health Physicist, who is assisted by his staff, with the support of the System Health Physicist and his staff. Gamma radiation levels in the pool area are constantly monitored by the station Area Radiation Monitoring System, which has a high level alarm feature. Additionally, periodic radiation and contamination surveys are conducted in work areas as necessary. Where there is a potential for significant airborne radionuclide concentrations, continuous air samplers are used in addition to periodic grab sampling. Personnel working in radiologically controlled areas must wear protective clothing and respiratory protective equipment, depending on work conditions, as required by the applicable Radiation Work Permit (RWP). Personnel monitoring equipment is assigned to and worn by all personnel in the work area. At a minimum, this equipment consists of a thermoluminescent dosimeter (TLD) and self-reading pocket dosimeter. Additional personnel monitoring equipment, such as extremity badges, will be worn by divers working in the pool.

Contamination control measures are used to protect persons from internal exposure to radioactive material, and to prevent the spread of contamination. Radiation Control Zones (RCZ's) are established around the work area. Work, personnel traffic, and the movement of material and equipment in and out of the area are controlled so as to minimize contamination problems. Material and equipment removed from the SFP will be rinsed, wiped down, decontaminated further if necessary, and wrapped and/or tagged as necessary. Divers exiting the pool water will also be rinsed off to minimize personnel and area contamination problems. A contamination containment tent with local filtered exhaust ventilation system is available for use in cutting up the removed rack sections if necessary and for placment into a shielded shipping cask. Good houskeeping habits are also practiced. The station radiation protection staff closely monitors and controls all aspects of the work to ensure that personnel exposures, both internal and external, are maintained as low as reasonably achievable (ALARA).

5.1.1 UNDERWATER RADIATION SURVEY

In addition to periodic measurement of dose rates around and above the pool, underwater surveys will be conducted to determine the dose rates in areas where divers must work or pass through. TLD's are used to perform these measurements. A low and high range underwater radiation monitoring instrument will be used, when applicable, to perform dose rate measurement underwater.

5.1.2 POOL DECONTAMINATION AND CLEAN-UP

The Spent Fuel Pool Cooling System provides purification and clarification of pool water by recirculating it through a demineralizer and filters. This system operates in this mode to minimize radiation exposure to personnel from the amount of dissolved and suspended radionuclides in the pool water. In addition, a portable filtered water vacuum system will be used, as necessary, to clean loosely deposited contaminants from the pool floor, walls, and fuel rack surfaces. This system has the dual purpose of (1) cleaning radioactive crud around diver working areas to minimize radiation exposures; and (2) removing metal particle residues from the pool from underwater cutting and grinding operations.

5.1.3 DIVING OPERATIONS

Prior to all diving operations, the spent fuel assemblies stored in the pool will be arranged in such a manner as to yield the lowest practicable dose rates to divers and still minimize the amount of movement or rearrangement of the assemblies. These two criteria should lessen the combined effects of radiation directly from the fuel, and radiation from activated particles in the water that are stirred up by fuel movement. Underwater travel paths will be established for divers to ensure that exposures received going to and from the work areas are maintained ALARA. Health Physics personnel will be in the immediate area at all times when divers are in the water. Their duties will be to provide health physics support to minimize personnel exposure, to enforce good radiological work practices and adherence to RWP requirements. They, along with the diver supervisor who will be in direct communication with the divers, will continually observe the divers while they are in the pool and ensure that the divers are warned if they approach high radiation/exclusion zones.

Divers will wear protective clothing items inside their rubber diving suits to protect them from contamination when they remove their diving suits and exit the SFP area. TLD's will be worn inside the diving suits on the chest, back, and extremities. Self-reading pocket dosimeters will be sealed in plastic bags and also worn inside the diving suit. The self-reading pocket dosimeters will be read after each dive and recorded. A daily tabulation of each individuals cumulative whole body and extremity doses will be prepared on each diver which will be reviewed by the diving supervisor and the cognizant Health Physics Supervisor. This information will be used in part (1) to maintain doses ALARA within the limits and (2) to efficiently allocate exposure among the divers working in the pools.

5.1.4 DECONTAMINATION OF REMOVED RACK SECTIONS

When rack sections are removed from the spent fuel pool, they will be rinsed with a low pressure spray using demineralized water or spent fuel pool water. Personnel involved in this operation or others in the immediate area will wear appropriate protective clothing and respiratory protective equipment. The rack sections will be allowed to drip dry prior to movement to the contamination containment tent in the new fuel receiving area for cut-up and/or packaging for disposal. This rinsing operation is expected to remove significant quantities of loose contamination from the racks while causing a relatively low exposure to decontamination personnel. This minimizes subsequent personnel exposures due to handling, cutting-up, and packaging of the rack sections for shipment and disposal.

5.1.5 REMOVED RACK SECTION CUT-UP AND DISPOSAL

After the rack sections are rinsed they will be moved to the new fuel receiving area and placed in a large contamination containment tent, which will then be closed. A localized high efficiency filtered exhaust ventilation system, which will be attached to the tent, will be employed at all times when work is being done in the tent to reduce airborne radionuclide concentrations and any smoke from cutting/burning operations. Fresh make-up air will be supplied to the tent. Special consideration is given to fire safety in the design and construction of the tent. If radiation levels warrant strict dose control, a fast cutting torch will be available to minimize time required to cut-up the rack into

sizes that permit efficient disposal. All involved personnel will wear appropriate protective clothing and respiratory protective equipment. The tent will not be opened for removal of the cut-up rack pieces until air monitoring shows that the tent ventilation system has reduced the interior airborne radionuclide concentration to acceptable levels. The tent interior will also be decontaminated as necessary to reduce the level of loose surface contamination to acceptable levels. This need for decontamination will be determined by periodic swipes (or smear) sampling. When conditions are acceptable, the cut-up rack pieces will be placed in the appropriate radioactive waste container/cask for shipping and subsequent disposal at an approved radwaste burial site.

5.2 RADIOLOGICAL EVALUATION

5.2.1 ANTICIPATED EXPOSURES DURING RE-RACKING

Table 5.2-1 is a summary of expected exposures for each phase of the re-racking operation and for each group of workers.

5.2.2.1 Solid Waste Generation

Presently the sources of solid radioactive wastes generated by the Spent Fuel Pool Cooling System consists of (1) approximately 21 $ft.^3$ per year of demineralizer resin, and (2) approximately 30 $ft.^3$ per year of spent filters, for an annual total of approximately 51 $ft.^3$. Conservatively estimating that the proposed increase in spent fuel storage capacity would necessitate changing filters and demineralizer resin at twice the present frequency, approximately another 51 $ft.^3$ of solid waste is expected to be generated, bringing the projected annual total to approximately 102 $ft.^3$.

5.2.2.2 Krypton -85

Data is not available regarding Krypton -85 measured from the spent fuel pool alone for the past two years. However, for the period from July 1, 1976 through June 30, 1978, a total of 637 Curies of Krypton -85 was released from the Oconee Nuclear Station as gaseous waste which was generated from all sources including all three units.

5.2.2.3 Anticipated Increases in Routine Personnel Exposures

The increases in doses to personnel from radionuclide concentrations in the spent fuel pool due to the expansion of the capacity of the spent fuel pool are evaluated as follows:

(a) Table 5.2-2 shows the most recent gamma isotopic analysis of spent fuel pool water, including the principal radionuclides present and their respective concentration. This analysis is conservative with respect to normal levels since it was taken immediately following the refueling of both Units 1 and 2.

(b) The dose equivalent rates around the pool perimeter and at the pool centerline/bridge level typically range 20-50 mrem/hour and 40-80 mrem/hour respectively during refueling periods. After a few weeks of cleanup using the Spent Fuel Pool Cooling System demineralizer and filters, dose equivalent rates typically range 10-20

mrem/hour around the pool perimeter and 20-40 mrem/hour at the pool centerline/bridge level during spent fuel shuffling in the SFP. Dose equivalent rates typically range 5-10 mrem/hour and 10-20 mrem/hour at these same locations, respectively, when no spent fuel activity is in progress. The variation in the dose equivalent rates are due to the varying amounts of loose crud introduced into the SFP from (1) transferring spent fuel assemblies and other components from the reactor to the SFP, (2) crud bursts from the core caused by operation of the low pressure injection system (decay heat removal system) during refueling operations and (3) periodic introduction of primary coolant make-up water into the SFP. The dose equivalent rate contribution from the spent fuel assemblies themselves is negligible.

(c) Table 5.2-3 shows an airborne radionuclide analysis of the spent fuel pool area, including the principal radionuclides present and their concentrations. This, too, is considered to be conservative with respect to normal conditions as noted in paragraph (a).

(d) The concentrations of airborne radionuclides in the SFP area are typically below 25% of the MPC values listed in the 10CFR20 Appendix B Tables. However, concentrations of certain radionuclides, such as I-131, at levels around 1 MPC have occurred during refueling periods. This is apparently due to moving defective fuel assemblies from the core into the pool. At times, introduction of primary coolant make-up water into the SFP has caused elevated airborne radioactivity levels. Therefore, only a negligible increase, if any at all, in the SFP work area or at the site boundary is expected due to the increased number of assemblies stored in the pool.

(e) It is conservatively estimated that the frequency of changing the Spent Fuel Pool Cooling System demineralizer resin and filters may double the annual person-rem burden of these operations. This represents a potential annual increase of approximately 3 person-rem.

(f) Crud build-up (e.g. Co-58, Co-60) along the sides of the pool has not been significant in the past. With the increase in spent fuel storage capacity, crud build-up may double. However, waterborne crud in the pool will be continually removed by the Spent Fuel Pool Cooling System demineralizer and filters. A filtered water vacuum is also available and will be used to clean the sides of the pool should radiation levels from this source become significant in the future. Therefore, radiation levels are not expected to increase significantly over a long time period.

(g) The total dose expected to be received annually by personnel occupying the spent fuel pool area based on all operations in that area after re-racking is complete, except for routine spent fuel shipments, is approximately 19 person-rem; a 15% increase. The doses due to fuel area work and those due to preparations for fuel and/or other shipments are expected to be reduced beyond that experienced to date as profficiency is gained in handling spent fuel assemblies, as well as spent fuel casks and associated equipment; and as better cask decontamination techniques are developed and used.

5.2.3 DISPOSITION OF OLD RACKS

The weight of the spent fuel racks to be removed from the SFP is approximately 23,000 pounds. Cut-up and disposal of the removed racks are discussed in Section 5.1.5.

Table 5.2-1

ANTICIPATED EXPOSURES DURING RE-RACKING
(person-rem)

	Initial Baseplate Survey	Rack Removal and Installation	Rack Cutting and Disposal	
Operations	7	22	---	
Construction				
Divers	2.5	18	---	
Others	---	2.5	15	
Health Physics	3.5	25	1.5	
Quality Assurance	---	12	---	
Engineering	1.5	6	---	
Janitorial	2	4	3	
				TOTAL
Total	16.5	89.5	19.5	125.5

TABLE 5.2-2

GAMMA ISOTOPIC ANALYSIS OF UNIT 1 AND 2 SFP WATER

RADIONUCLIDE	CONCENTRATION (μC/ml)
Ar-41	6.032×10^{-4}
Kr-85m	3.399×10^{-3}
Kr-87	3.137×10^{-3}
Kr-88	5.515×10^{-3}
Xe-133	1.221×10^{-1}
Xe-133m	3.297×10^{-3}
Xe-135	3.696×10^{-2}
Ma-24	3.005×10^{-4}
Mn-54	7.547×10^{-4}
Co-58	5.171×10^{-4}
I-131	6.128×10^{-3}
I-132	6.182×10^{-3}
I-133	7.465×10^{-3}
I-134	7.798×10^{-3}
I-135	6.489×10^{-3}
Sr-91	8.222×10^{-4}
Sr-92	1.208×10^{-3}
Ru-106	2.044×10^{-3}
Cs-138	2.056×10^{-2}
Ba-140	9.560×10^{-5}

TABLE 5.2-3

GAMMA ISOTOPIC ANALYSIS OF AIRBORNE RADIONUCLIDES
IN THE UNIT 1 AND 2 SFP BUILDING

RADIONUCLIDE	CONCENTRATION (μC/ml)
I-131	4.8×10^{-10}
Cr-51	5.4×10^{-11}
Co-58	1.9×10^{-10}
Co-60	9.2×10^{-12}
Ag-110m	3.5×10^{-12}
Sr-92	1.5×19^{-12}
Ru-103	2.1×10^{-11}
Cs-134	1.3×10^{-12}
Cs-137	6.4×10^{-11}

6.0 SAFETY ANALYSIS

The following analyses are provided to update and reverify postulated accidents associated with operations in and around the spent fuel pool.

6.1 CASK/HEAVY-LOAD DROP ACCIDENT

In order to calculate the consequences of a cask drop accident, it is necessary to determine the maximum number of fuel assemblies which could be contacted. The worst case is considered to be a hoist cable failure when the cask is positioned over the fuel pool wall and the cask has an eccentric drop onto the wall. In this case, the cask, yoke and load block could be deflected onto the spent fuel.

There are 68 cans under the projected cask yoke and block impact area. These cans buckle and deflect into adjacent cans until the total energy of the falling cask is absorbed. In total, 205 cans can potentially suffer a total loss of integrity during a cask drop accident.

The radiological consequences of the cask drop accident will be mitigated by limiting the age of fuel stored in first 28 rows. No cask movement will be allowed if fuel in these locations has decayed less than 50 days. The worst radiological consequences experienced would result from 100% of the activity contained in the fission gases trapped in gaps in the fuel stored in the 205 locations being released into the pool water. The exclusion area boundary dose, taking no credit for ventilation system filtration, would be 2.8 rem whole body and 127 rem to the thyroid. These doses are well below 10CFR Part 100 limits.

6.2 CONSTRUCTION ACCIDENT

The approximately 140 fuel assemblies in the fuel pool during the installation of the new racks, will have decayed at least 150 days prior to reracking. Therefore, the possibility of the failure of a temporary crane or other construction accident is considered less severe than a cask drop accident. As an additional precaution, the rack modules will not be moved over stored spent fuel during the installation process.

6.3 FUEL HANDLING ACCIDENT

The potential for an accident during fuel handling operations is addressed in previous submittals as well as Section 14.2.2.1 of the Oconee FSAR. The increased capacity does not affect the severity of fuel handling accident. Thus, previous analyses and conclusions continue to be valid.

6.4 LOSS OF FORCED COOLING

The large volume of water in the spent fuel pool takes several hours to heat up to boiling if all cooling capacity is lost. There is ample time to effect repairs to the cooling system or arrange alternate cooling should adequate cooling capacity be lost. The amount of time before the pool begins to boil is dependent on both the heat load and the initial pool temperature. The time to adiabatically heat up to boiling from the normal operating temperature of 125°F given in the FSAR and from the maximum temperature of 150°F is shown in Table 6.4-1.

Table 6.4-1

TIME BEFORE BOILING WITHOUT FORCED COOLING

Heat Load (10^6 BTU/hr)	Initial Pool Temperature (°F)	Time Before Boiling (hrs)
19.6	125°F	17.2
19.6	150°F	11.8
31.7	125°F	10.6
31.7	150°F	7.3

7.0 CONSIDERATION OF ALTERNATIVES

7.1 FEASIBILITY AND DESCRIPTION OF ALTERNATIVE ACTIONS

As indicated in Section 1, the purpose of this proposed action is to temporarily resolve the adverse conditions with regard to "back-end" of fuel cycle, caused by recent reversals in national policy. Along with the proposed action, other alternatives have been considered. The feasibility of each is discussed below.

7.1.1 RERACKING THE OCONEE 1 & 2 SPENT FUEL POOL

The expansion of on-site storage facilities can be achieved through the installation of high density storage racks. This alternative, one of the two proposed by Duke, is discussed fully in the preceding sections and is considered acceptable.

7.1.2 UTILIZATION OF REPROCESSING FACILITIES

Reprocessing facilities could be utilized to alleviate the spent fuel storage problem in two ways. The preferrable alternative is to use the facility for its intended purpose. It, however, has been eliminated by President Carter's announcement which, in the short term, ended consideration of licensing activities directed toward restarting the West Valley plant or starting the Barnwell facility.

The other option is to utilize the reprocessing plant and its facilities as an away-from reactor (AFR) storage facility. Such use, at this time, is not feasible for Oconee. Additionally, Barnwell is not licensed for either activity presently, and it is not anticipated that such licenses can or will be received in the near future. This option would not be undesirable, if available. The availability of this option will remain in question until the Department of Energy, the Nuclear Regulatory Commission and Allied come to terms on the future of Barnwell.

7.1.3 INDEPENDENT SPENT FUEL STORAGE FACILITY

The construction of an Independent Spent Fuel Storage Facility (ISFSF) by Duke is also a plausible temporary solution to the shortage of storage space. The main consideration, other than costs addressed in 7.2, is the time involved. The design, licensing and construction of an ISFSF have been estimated to take some 45 months. This estimate is based on a water-basin type storage facility which could only be licensed for 10 years (proposed 10 CFR 72). There is a great deal of uncertainty in such estimates considering this would be a first-time project. The actual technology involved is relatively simple and well understood but each licensing action involved would be precedent setting and would be extremely susceptible to delay. With regard to Duke's particular situation, this alternative would be untimely to the extent of being infeasible. It should be further noted that the location of an ISFSF (at Oconee, or other Duke-owned reactor site) would not have a significant impact on the feasibility or costs.

Another facet of the ISFSF alternative is to consider on ISFSF constructed by an entity other than Duke (e.g., DOE or TVA). There is considerably greater uncertainty in this alternative. Someone, other than Duke, who might not appreciate the great need for expeditious design/construction/licensing to prevent cessation of Oconee operations, would, in all likelihood, take longer to get such a plant into operation. Simply put, this alternative would have all the disadvantages of a Duke-built ISFSF to an even greater degree.

7.1.4 TRANSSHIPMENT

The possible use of an already constructed storage pool at a site other than Oconee is available. The shipment of Oconee spent fuel to McGuire is considered to be the best solution, albeit a temporary one, to the correct shortage of storage space at Oconee. The McGuire Unit 1 spent fuel pool will be available for storage of spent fuel in 1979. The McGuire Unit 2 and Catawba pools (which have been expanded for this purpose) will be available sometime in the early 1980's. As discussed in Section 1, this alternative is currently being pursued by Duke.

7.1.5 CESSATION OF OCONEE OPERATIONS

The remaining 'alternative' is to choose to not solve the problem. That is, if production of spent fuel ceases, the problem will appear to be resolved. This presents a totally unacceptable alternative. Oconee Nuclear Station represents in excess of 20% of the generating capacity of the Duke system. Duke does not possess sufficient capacity to sustain the complete loss of Oconee capacity and still meet the service demands. There are no sources of power which could be purchased except on an emergency basis.

The possibility of using Oconee on a last-on/first-off or other reduced power basis would only delay the inevitable and would be a complete misuse of resources committed to the construction of the Oconee station. Oconee is designed as a base load plant. The accelerated degradation of the vessel and other components which would accompany such cyclic utilization would promote the unwarranted waste of a considerable portion of the resources, both financial and natural, which make up the Oconee Nuclear Station. Thus, this 'alternative' is unacceptable.

7.2 COST/BENEFIT APPRAISAL

The alternatives, discussed in Section 7.1, are compared below on several bases. The costs include varying degrees of uncertainty but represent best-estimate costs and have been compared with other cost estimates as referenced.

7.2.1 COMMITMENT OF FINANCES

Alternative	Cost
Rerack Oconee 1-2 pool	\$ 8,300 per location
Utilization of reprocessing facility[1] (storage)	48,000 per location (if available)

Alternative	Cost
Construction of an ISFSF	$ 34,500 per location
Transshipment to McGuire	2,100 per assembly
Cessation of Oconee operations	$\geq$635,000 per day

[1]Omaha Public Power District (Ft. Calhoun) submittal on re-racking dated April 19, 1976 (escalated at 8% per annum)

The apparent conclusion on a cost-only basis is that only two alternatives are available at a "reasonable" cost; they are re-racking the Oconee 1 & 2 pool and shipment to McGuire. Both of these alternatives would require additional actions which would include further costs. These further costs would not approach the costs of the other alternatives.

7.2.2 COMMITMENT OF NATURAL RESOURCES

Tabulated below is a summary of the degree to which resources would be committed in order to carry out the various options.

Alternative	Resource Commitment
Rerack Oconee 1-2 pool	Resource commitment would be limited to structural steel used in the Hi-Cap racks and the land area necessary for disposal of the old racks.
Utilization of Reprocessing Facility for Storage	Utilization of space of reprocessing plant would require expansion of storage pool (re-racking or physical expansion). Therefore, commitment would be no less than expanding Oconee pool.
Construction of ISFSF	Construction materials (concrete steel, etc.) would obviously require a significantly larger commitment of natural resources than would reracking the Oconee 1 & 2 pool.
Transshipment	The only resources utilized in the transshipment option would be those used in construction of casks and in fuel, etc., used in transport.
Cessation of Oconee Operations	The costs in 7.1 are based on using fossil capacity to replace Oconee (such capacity is <u>not</u> available). Thus, this option would, as a minimum, require the use of large sums of coal or petroleum products. The increased use of either would be con-

trary to national energy policy and would require a huge commitment of resources. To actually carry out this option, it would be necessary to build substitute generating capacity which would require resources several orders of magnitude larger than the options chosen.

8.0 PROPOSED FINDINGS/CONCLUSIONS

In Federal Register Notice (FR 42801) dated September 10, 1975, the Commission announced its intent to publish a Generic Environmental Impact Statement on Handling and Storage of Spent LWR Fuel the draft of which was published in March, 1978. In the interim, the NRC proposed five criteria by which to judge the impact of licensing decisions in this regard. It is considered that this action falls well within that guidance. Section 8.1 below speaks to those considerations. Section 8.2 generally summarizes the conclusions reached in previous sections of this report.

8.1 CONSIDERATION OF FR 42801

(1) "It is likely that each individual licensing action of this type would have a utility that is independent of the utility of other licensing actions of this type."

The licensing actions proposed by Duke, transshipment and reracking the Oconee 1 and 2 pool are, if taken as one action, independent of the utility of any other possible actions. The two actions viewed separately do address the same concern. It should be noted, as it was in the "Environmental Impact Appraisal Related to Spent Fuel Storage of Oconee Spent Fuel at McGuire Unit 1" published by NRC/ONMSS in December of 1978, that transshipment alone might not and indeed probably would not be sufficient to ameliorate the shortage of storage space. It was suggested therein and is proposed hereby that the reracking option be pursued in addition to transshipment. The basic fact remaining is that neither option is self-sufficient or capable of completely solving the problem. Additionally, the propriety of proposing both actions in somewhat concurrent time frames is justified considering ALARA exposure levels and the maintenance of some flexibiity to absorb delays in licensing, engineering and/or construction. The utility of either or both actions is not dependent on the utility of other options.

(2) "It is not likely that the taking of any particular licensing action of this type during the time frame under consideration would constitute a commitment of resources that would tend to significantly foreclose the alternatives available with respect to any other licensing action of this type."

As discussed in Section 7.2.2 the resource commitments for the reracking option are not significant and are not of such magnitude to foreclose other alternatives. Indeed, the other alternatives discussed (expansion of reprocessing plant storage, ISFSF, and cessation of Oconee operations) would require a considerably greater commitment of resources. Only the reprocessing plant option would be comparable in this regard.

(3) "It is likely that any environmental impacts associated with any individual licensing actions of this type would be such that they could adequately be addressed within the context of the individual licensing action without overlooking any cumulative environmental impacts."

There are no environmental effects resulting from this action which have not or could not be adequately quantified and analyzed.

(4) "It is likely that any technical issues that may arise in the course of a review of an individual licensing application can be resolved within that context."

It is considered that most, if not all, of the technical issues which could arise during the review of this application are already adequately addressed in this submittal. However, if questions of a technical nature arise during the review, they will be answered in an expeditious and responsive manner. Any questions will be adequately resolved before the issuance of the SER which would accompany approval of this request.

(5) "A deferral or severe restriction on licensing actions of this type would result in substantial harm to the public interest..."

In Section 7.1 the impact of cessation of Oconee operations, which would be the result of deferral or severe restriction is discussed. The basic concern is to maintain an adequate reliable source of electric power for the region. This would be virtually impossible without Oconee generating capacity.

8.2 CONCLUSION

It is considered that the information presented in this document is sufficient to support the granting of NRC approval with regard to reracking the Oconee Unit 1 and 2 Spent Fuel Pool.