REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:79042502[illegible] DOC.DATE: 79/04/20 NOTARIZED: NO DOCKET #
FACIL:50-269 OCONEE NUCLEAR STATION, UNIT 1, DUKE POWER CO. 05000269
50-270 OCONEE NUCLEAR STATION, UNIT 2, DUKE POWER CO. 05000270

AUTH.NAME	AUTHOR AFFILIATION
PARKER,W.O.	DUKE POWER CO.
RECIP.NAME	RECIPIENT AFFILIATION
DENTON,H.R.	OFFICE OF NUCLEAR REACTOR REGULATION

SUBJECT: FORWARDS ADDL INFO RE STRUCTURAL ANALYSIS & RADIOLOGICAL EFFECTS OF PROPOSED INSTALLATION OF HIGH CAPACITY SPENT FUEL RACKS IN POOL.INCLUDES QUESTION RESPONSES,NON PROPRIETARY DRAWINGS,& ADDL TECH SPEC CHANGE.

Smf Pp75

DISTRIBUTION CODE: A001S COPIES RECEIVED:LTR 40 ENCL 40 SIZE: 48
TITLE: GENERAL DISTRIBUTION FOR AFTER ISSUANCE OF OPERATING LIC

NOTES: M CUNNINGHAM - All AMDTS TO FSAR + CHANGES TO TECH SPECS.

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
ACTION:	05 BC ORB 4	7	7			
INTERNAL:	01 REG FILE	1	1	02 NRC PDR	1	1
	12 I&E	2	2	14 TA/EDO	1	1
	15 CORE PERF BR	1	1	16 AD SYS/PROJ	1	1
	17 ENGR BR	1	1	18 REAC SFTY BR	1	1
	19 PLANT SYS BR	1	1	20 EEB	1	1
	21 EFLT TRT SYS	1	1	22 BRINKMAN	1	1
EXTERNAL:	03 LPDR	1	1	04 NSIC	1	1
	23 ACRS	16	16			

TJA2

APR 26 1979

TOTAL NUMBER OF COPIES REQUIRED: LTTR 38 ENCL 38

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

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

April 20, 1979

TELEPHONE: AREA 704
373-4083

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 Reactors Branch #4

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

Dear Sir:

My letter of February 2, 1979 requested approval of an amendment to the Oconee Nuclear Station Technical Specifications which would license the installation of high capacity spent fuel racks in the shared Oconee 1, 2 pool. Your letters of March 13, 1979 and March 29, 1979 requested additional information in the area of structural analysis and radiological effects of the proposed re-racking. A total of thirty-one (31) questions have been received thus far. The responses to these questions are attached in the following manner:

Attachment 1	Question Responses
Attachment 2	Non-Proprietary Drawings
Attachment 3	Additional Technical Specification Change

The information provided is considered to adequately respond to the concerns raised and should allow the review of the modification to be completed in an expeditious manner. The information has also been discussed with various members of the Staff.

The Technical Specification request included herein is a supplement to the February 2, 1979 submittal and as such no license fee is attached.

Very truly yours,

William O. Parker, Jr.

KRW:scs
Attachments

A001
S
40/40

7904250203

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; 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 20th day of April, 1979.

Vivian B. Hopkins

Notary Public

My Commission Expires:

February 15, 1982

ATTACHMENT 1

QUESTION RESPONSES

1. Question

Provide the ratio of horizontal displacement, due to the SSE, and the availability gaps between racks and between racks and nearest pool structure. This ratio of displacement to gap is the margin against contact during the SSE, which would result in an unanalyzed force. This question is referenced to your February 2, 1979 letter, Attachment 2, page 2-4.

Response

Based on nonlinear, time history, sliding base analyses performed as stated in the February 2, 1979 submittal, page 2-4, the ratios of horizontal displacement to the minimum available gaps between adjacent racks and between the racks and nearest spent fuel pool structure are less than 0.11 and 0.22 respectively. The actual sliding distance will not exceed 0.133 inches.

2. Question

Describe the positive physical means you will employ to maintain a potential drop height of nor more than 6 feet for a fuel bundle. This question is referenced to page 2-5.

Response

The potential for dropping a fuel assembly could occur only during two situations, both of which would result in a drop of considerably less than the six feet analyzed.

The most routine situation is the transport of fuel to/from the racks during refueling or other fuel movement operations. The maximum drop height from the "full-up" position on the fuel masts is around four feet.

The other situation involves an accident during fuel movement associated with post-irradiation examination (PIE). The PIE equipment utilizes a jib crane to move the fuel between the racks and the examination frames. The grappling device is attached to a solid bar which is, in turn, attached to a cable. The fuel is raised/lowered by winding the cable around a hoist drum. At the point when the bar reaches the drum, the maximum drop height is four feet.

3. Question

Verify that all provisions of the NRC guidance on spent fuel pool modifications, entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications" (including errata), regarding the use of ASME Code, Subsection NF are met. Justify any deviations.

Response

All provisions of the "Review and Acceptance of Spent Fuel Storage and Handling Applications" (including errata) regarding use of ASME Subsection NF are met to the extent delineated in each analysis. Specifically, the new rack modules are designed to meet all ASME NF criteria which are applicable and the pool continues to conform to FSAR analysis methodology and acceptance criteria.

4. Question

(Page 22) Provide a more detailed sketch of typical module than p.8, 2.2-2. Include details of "chevron grid structure," supporting "U" channels, connections between various elements. Provide dimensions, thicknesses, size of gaps and in general all the information needed to evaluate strength of elements. Make clear how the modules are supported.

Response

The drawings requested are provided in Attachment 2.

5. Question

(Page 2-2 and 2-3) It is stated that non-linear time history method of seismic analysis has been used for two horizontal directions and the response spectrum method for the vertical direction, also that these methods represent the response of the pool structure to the specified ground motion. Clarify this statement. Explain, step by step, the method used to introduce as input the response of the pool. Justify the compatibility of two different methods (time history and response spectrum) used for the same structure.

Response

The words "time history" and "response spectrum" do not refer to the methods of analysis used to determine the pool floor response. The pool floor response histories were calculated as described on page 2-3. The pool floor lateral histories were then used to perform lateral non-linear time history analyses of the racks and the pool floor vertical history was converted to a response spectrum for use in a vertical linear response spectrum analysis. The use of non-linear time history analyses in the two lateral directions was required by the non-linear characteristics of the fuel racks in those directions. Due to the linear nature of the fuel racks in the vertical direction, only a linear response spectrum analysis was required in the vertical direction. The methods of analyzing and combining response loads for the fuel racks in the three component directions is consistent with the requirements of NRC Regulatory Guide 1.92.

6. Question

You state (page 2-3) that first an analysis was performed assuming an empty rack module, that from this a dynamically equivalent model of the module was determined, and used in the final analysis of the rack containing fuel assemblies, and including the effects of water inside and outside of the module. Explain with precision the transition from one model to the other.

Response

As discussed on page 2-3, the spent fuel rack dynamic analysis is performed in two steps. The first step is to determine the dynamic characteristics (natural frequency and mode shape) of the rack module itself from a detailed, finite element model (Figure 2.3-1). The second step determines the dynamic response of an equivalent rack model (Figure 2.3-2) to the effects of fuel impacting, hydrodynamic action and the acceleration time history of the pool floor. The transition from step 1 to step 2 is as follows: an examination of the mode shape of the rack module indicated that the lateral deflection varies primarily along the height of the rack, remaining essentially constant in any horizontal plane. Therefore, the model was reduced to a simple vertical array of springs and masses which duplicates the weight, natural frequency and mode shape of the original 3-dimensional model. The mass modes are allowed both translational and rotational degrees of freedom and because of the importance of fuel impacting, are located at the same elevations as the fuel spacer grids. Before this model could be used in CESHOCK, a fuel model was added as were special elements to duplicate the effect of impacting between the two. Other elements duplicated the hydrodynamic mass and coupling among the fuel, racks, and pool walls; friction among the fuel, racks and pool floor and rocking of the modules on their supports.

7. Question

You indicate that the SAP and the CESHOCK computer codes have been used. Explain what steps have been taken to insure that the inputs were checked, also the results. Did you comply with the requirements of Standard Review Plan 3.8.1 Section II.4.e?

Response

The analysis including computer input and results are verified in accordance with the quality assurance procedure contained in CENPD 210 A Rev. 3, "QA Manual for Nuclear Steam Supply Systems", November 1977. This procedure complies with the requirements of Standard Review Plan 3.8.1, Section II.4.e.

8. Question

(Page 24) You do not provide details of the base of the module. Provide a sketch and explain in detail the path along which the forces are transmitted to the bottom of the pool.

Response

The "base" of the modules and the path over which the loads are transmitted are shown in sketches/drawings included in Attachment 2. To clarify the path, the following description is provided.

a) Figure 4 of Attachment 2 shows the general construction of a typical spent fuel storage rack module. In this design, chevron beams are strategically located at two elevations. Individual storage cavities are then structurally connected to these chevron beams in the manner shown in section G-G of Figure 4. The entire spent fuel rack module is supported by four channels. These channels are connected to the lower ends of eighteen (18) cavities along the outer periphery of the module. Detail of this construction is illustrated in Figure 5. During a postulated seismic event, dynamic loads developed by interior storage cavities are transmitted outward toward these peripheral cavities via the chevron grid structures. Total module load will then be transmitted to the supporting channels and through the bearing pads to the pool floor.

b) The "U" channels sit on 12 bearing pads per module as shown in Figures 1 and 5. These pads are not attached to the "U" channels or the pool liner. They serve to elevate the modules above the "bent plates" (also in Attachment 2) which form part of the existing rack structure. By including these pads it alleviates the need to grind the existing structure flush with the liner thereby improving ALARA dose levels. Sufficient area is provided to give necessary frictional contact at both the module/pad and pad/liner interfaces.

9. Question

Discuss the effect of increased vertical and horizontal loads on the liner and on the concrete structure (walls and bottom) of the pool.

Response

The fuel pool floor was reanalyzed taking into account any additional loads resulting from the proposed increase in pool storage capacity. Seismic loadings were based on preliminary rack pool interface loading data from Combustion Engineering's analysis procedure as described in Section 2.3.1 of the February 2, 1979 submittal. The preliminary loadings were conservatively increased by 10% in all cases to account for any possible variation between preliminary and final rack-pool interfacing loadings. Final rack-pool interface loadings will be compared to actual loadings used to insure that they were conservative in all cases.

As stated in Section 3.1 of the submittal, Oconee FSAR Appendix 5A design criteria was utilized for the reanalysis of existing structures. The following table presents a comparison of the most critical calculated moments as defined by the most severe loading combinations and the allowable moments. Comparison of the computed values with the allowable values shows the fuel pool floor is adequate to withstand the new racks and additional fuel.

SUMMARY OF STRUCTURAL EVALUATION FOR
ADDITION FLOOR LOADINGS

	Loading Condition	Critical Load Combination	Moment	
			Allowed	Calculated
Pool Slab $\frac{\text{ft-kips}}{\text{ft}}$	Normal (AC1-318-63)	1.5D+1.8L+1.5To	618 $\frac{\text{K-ft}}{\text{ft}}$	221 $\frac{\text{K-ft}}{\text{ft}}$
	Accident & Seismic	1.0D+1.0E'+1.0Ta	618 $\frac{\text{K-ft}}{\text{ft}}$	286 $\frac{\text{K-ft}}{\text{ft}}$
Pool Floor Stiffening Members ft-kips	Normal (AC1-318-63)	1.5D+1.8L+1.5To	4700 K-ft	3016 K-ft
	Accident & Seismic	1.0D+1.0E'+1.0Ta	4700 K-ft	3270 K-ft

Ta - As defined in Section 3.1 of the submittal

To - As defined in Section 5.7.1.2 of the FSAR

D - Dead load

E' - Horizontal and vertical seismic loads as described in Section 2.3.1 of the submittal and explained above.

L - Live load

9. Response (cont'd)

The new rack modules are free standing and are not supported in any way off the pool walls. The response to Question 1 indicates that there are also no impact loadings on the pool wall during a seismic event.

A reanalysis was also conducted to determine the effect of increased vertical and horizontal loads on the liner. The resultant stresses from the maximum lateral seismic forces exerted by the rack modules on the pool floor were combined with the thermal stresses in the floor liner (i.e., the most critical combination of 1.0D+1.0E'+1.0Ta). For resistance to seismic forces, the thickness of the liner plate, exclusive of the stainless steel cladding, was used. The presence of the cladding was considered, however, with respect to thermal effects. The following analysis results show that the maximum stresses in the liner and the welds connecting the liner to embedments in the concrete are below normal design allowables:

	Computed Stress	Normal Allowable Stress
Liner Plate	9.7 ksi	21.6 ksi
Welds	17.6 ksi	24 ksi

10. Question

Most of the published studies on hydrodynamic affects discuss unidirectional motion only. The present problem involves two-directional even tri-directional motion. Since the principle of superposition is not applicable to non-linear systems, your analysis is only approximation. Therefore justify your approach to this problem, and establish that it offers reliable results.

Response

According to the NRC guidance on spent fuel pool modifications of April 14, 1978 entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications", the recommendations provided in the publication, "Effective Mass and Damping of Submerged Structures", (UCRL-52342) by R. G. Dong, provide an acceptable basis for staff review. The methods of accounting for hydrodynamic effects in the fuel rack analysis are encompassed by the recommendations of the above reference.

11. Question

Describe the planned inservice inspection and surveillance of the new racks and liner supporting them. On page 3-1 you indicate that the pool liner is constructed from ¼" stainless steel clad plates. Discuss the potential for corrosion of the liner, after miscellaneous dynamic and thermal loads occurring during operation of the plant (including earthquakes) initiate some defects in the cladding.

Response

The inservice inspection of spent fuel pools is not currently required of the Oconee Nuclear Station nor, to the best of our knowledge is it required of licensees in general. The only "inspection" anticipated is any visual inspection which would seem reasonably prudent during diving operations in the pool during installation of the new racks. In response to question 1 it was determined that the maximum sliding expected even during DBE seismic events is approximately 0.1". It is considered reasonable to conclude that any movement which might result from a credible event would not be likely to introduce signficant damage to the pool liner cladding. There does not appear to be justification for the imposition of any inservice inspection requirements.

12. <u>Question</u>

Provide Figure 4.2.1 referred to on page 4-1.

<u>Response</u>

Page 4.1 contains a typographical error, it should read Figure 4.1-1 which is included.

13. Question

(Page 4-1) You state that "...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..." Provide sketches explaining this operation.

Response

Figures 2 and 3 of Attachment 2 provides sketches to show the general layout of typical existing rack structures and the locations of expected cuts for the removal operation.

14. Question

On same page (4-1) you state that you will avoid transporting new rack modules above stored fuel. Indicate the procedure which will enforce this requirement.

Response

The path over which the modules are to travel will be administratively controlled to preclude movement over stored fuel. However, no unanalyzed accidents could occur even if the module were somehow dropped onto stored spent fuel. The accident analysis presented in Section 6.1 and 6.2 show how the drop of any heavy load or other construction accidents are conservatively enveloped by the cask drop analysis and resulting limitations on fuel age.

Additionally, a proposed Technical Specification change, requested by the staff in Question 31 will prohibit transportation of any suspended loads of more than 3000 lb_m [the weight of fuel assembly, control rod and handling tool(s)] over stored spent fuel.

15. Question

On same page (4-1) you also refer to "acceptable" tolerances on module verticality, levelness and positioning. List these tolerances and describe the means of enforcement.

Response

Acceptable tolerances are based on fuel handling equipment accessibility. This will be verified on a go/no-go basis by inserting dummy assemblies in every fuel cavity.

16. Question

On page 4-2 you refer to "bearing pads". Provide a sketch of typical pad and its connections to the module and to the liner if any.

Response

Figure 1 of Attachment 2 provides the requested sketch. Please refer to response to Question 8 for further explanation.

17. <u>Question</u>

On page 4-3 you refer to "bent plates" supporting the old racks and welded to the liner. Sketch these bent plates, indicate their relationship to the "pipes" mentioned in our question (13) above and describe the method of removal of these bent plates if required.

<u>Response</u>

Figures 2 and 3 of Attachment 2 provide requested sketches. The bent plates are welded to the liner to form the bottom of the fuel cavities as shown in Figure 3. The pipes form the vertical structure of the "racks" themselves. In that the current racks are not of the can-type design, there are no cavities as such. The "pipes" and "angles" form a framework which support and connect the upper and lower portions of the racks.

18. Question

On page 4-3 you refer to a "temporary construction crane". Describe protection against drop of a heavy load such as proposed in Question 31.

Response

Please refer to Response to Question 14.

19. <u>Question</u>

There will be a time interval during which old and new racks will be in place simultaneously. Indicate that a sufficient physical separation will be provided to avoid interaction.

<u>Response</u>

The expected sliding of the new racks under seismic excitation (see response to Question 1) is approximately 0.1 inches. The gaps between new modules and old rack structure during installation is on the order of feet. Specifically the following gaps are present.

Step (See Section 4.1)	Distance (approx. feet) Between Old Racks and Modules	Distance (approx. feet) Between Fuel in Old Racks and Modules
1	NA	NA
2	NA	NA
3	5'	20'
4	5'	30'
5	15'	30'
6	15'	25'
7	5'	15'
8	5'	30'
9	15'	30'
10	15'	25'
11	5'	15'
12	5'	-
13	(all old racks removed)	(all fuel in new racks)

Therefore, sufficient physical separation is provided to avoid interaction.

20. Question

Figure 2.3-2, describe the properties of the spring element between the bottom lumped mass and the fixed base used in the vertical direction analysis.

Response

The model shown in Figure 2.3-2 was only used for the nonlinear lateral analyses. The linear vertical analysis was based upon the model shown in Figure 2.3-1 with the effects of fuel loading and water taken into account.

21. Question

Page 4-2, verify that during installation of the new storage modules, a drop of a module will not damage the spent fuel pool liner. Provide the maximum module drop height and worst case module drop orientation.

Response

The drop of a module during construction/installation could damage the spent fuel pool liner. However, the resulting damage is considered to be conservatively enveloped by the cask-drop analysis presented to the Commission in support of amendments 32/32/29 to licenses DPR-38,-47,-55 respectively, reference Enclosure 4 (SER and EIA) to Mr. A. Schwencer's letter of September 10, 1976 issuing the above referenced amendments.

22. Question

You stated in your February 2, 1979 submittal that the dose rates above the spent fuel pool are expected to be 5 to 10 mrem/hour at times other than refueling. Justify why this radiation field will be as low as reasonable achievable (ALARA) during the proposed pool modification. Your response should consider increased purification system operation, increased filter and demineralizer change out frequency or relocating tools and components stored in the pool.

Response

The primary causes of the relatively high equilibrium dose rates presently observed in the pool area are considered to be failed fuel, inpool testing and the effects of 9 refuelings. The dose rates in the Unit 1 and 2 Spent Fuel Pool area are expected to be 5 to 20 mrem/hour during the proposed pool modification. Present dose rates are in the range of <5 to 14 mrem/hour; however, past experience has proven that dose rates in the pool area are, to some degree, related to the movement of fuel in the pool. During the course of the proposed modification, individual spent fuel assemblies will be shuffled back and forth frequently (an estimated 436 times). It is anticipated that this fuel movement should increase dose rates in the pool area to the expected 5 to 20 mrem/hour.

All practicable methods are employed to maintain radiation levels from the pool as low as reasonably achievable. The Spent Fuel Pool Cooling System filters and demineralizers are utilized to clean up pool water at their available capacity. All identified leaking spent fuel assemblies have been transferred to the Unit 3 Spent Fuel Pool as a radiation source reduction measure. All extraneous tools, components, and testing equipment will have been removed from the pool or shielded prior to modification work commencing in the pool area. The pool floor will be thoroughly vacuumed before work begins, as will any other underwater surfaces where such vacuuming will result in a significant reduction in working area dose rates. Low exposure waiting areas and travel paths will be designated to minimize worker doses. These principal methods, along with lesser measures, ensure that pool modification dose rates will be ALARA.

23. <u>Question</u>

In accordance with Table 5.2-1 of the aforementioned submittal, the occupational exposure expected for the spent fuel pool modification is estimated to be 125.5 man-rem. Provide the data showing the derivation of this estimation. The data should include the expected dose rate to workers during each phase of the operation, the number of people involved and their occupancy times. Include the exposure that will be received from removal, decontamination and disposal of miscellaneous equipment presently stored in the pool.

<u>Response</u>

Since the original estimate of 125.5 man-rem for completion of the proposed modification was submitted, a subsequent estimate has been calculated using more reliable data and additional information which was unavailable previously. Also, information from the Ginna modification was reviewed and factored into the revised estimate where applicable. The total occupational exposure necessary to accomplish the pool modification is presently estimated to be approximately 76 man-rem. This estimate is broken down in the revised Table 5.2-1 as to work group, number of individuals involved, occupancy time, average dose rate, and job exposures. It should be noted that uncertainties exist as to the effectiveness of underwater vacuuming in reducing dose rates to divers, and as to radiation levels from removed rack sections.

TABLE 5.2-1 (revised)

OCONEE NUCLEAR STATION

SPENT FUEL POOL 1 & 2 MODIFICATION DOSE ESTIMATES

1. Removal, Decontamination, and Disposal of Miscellaneous Equipment Presently Stored in Pool:

Work Group	No. of Individuals	Occupancy Time (Man-Hrs.)	Avg. Dose Rate (mrem/hr)	Job Exposure (Man-Rem)
Operations	2	70	10	0.7
Engineering	2	70	10	0.7
Total	4	140	-	1.4

2. Underwater Vacuuming:

Work Group	No. of Individuals	Occupancy Time (Man-Hrs.)	Avg. Dose Rate (mrem/hr)	Job Exposure (Man-Rem)
Miscellaneous[1]	2	160	15	2.4
Operations	1	80	10	0.8
Health Physics	1	80	5	0.4
Total	4	320	-	3.6

3. Base Plate Survey:

Work Group	No. of Individuals	Occupancy Time (Man-Hrs.)	Avg. Dose Rate (mrem/hr)	Job Exposure (Man-Rem)
Miscellaneous				
diver	1	30	100	3.0
diver supvr.	1	30	10	0.3
Engineering	5	20	10	0.2
Health Physics	1	30	10	0.3
Janitorial	2	10	10	0.1
Total	10	120	-	3.9

4. Rack Removal and Installation:

Work Group	No. of Individuals	Occupancy Time (Man-Hrs.)	Avg. Dose Rate (mrem/hr)	Job Exposure (Man-Rem)
Miscellaneous				
divers(underwater)	5	300	100	30.0
divers(pool side)	5	300	5	1.5
diving supvr.	1	300	10	3.0
Operations				
bridge operators	2	1200	10	12.0
crane operators	2	800	5	4.0
Engineering	2	200	5	1.0
Health Physics	3	1200	10	12.0
Quality Assurance	1	20	10	0.2
Janitorial	2	200	5	1.0
Total	23	4520	-	64.7

5. Rack Disposal:

Work Group	No. of Individuals	Occupancy Time (Man-Hrs.)	Avg. Dose Rate (mrem/hr)	Job Exposure (Man-Rem)
Miscellaneous[2]	2	18	100	1.8
Health Physics	2	14	5	0.1
Janitorial	2	14	5	0.1
Total	6	46	-	2.0

GRAND TOTALS

Work Group	No. of Individuals	Occupancy Time (Man-Hrs.)	Effective Dose Rate (mrem/hr)	Job Exposure (Man-Rem)
ALL	47	5146	15	75.6

[1]Vendor underwater vacuum operators

[2]Vendor power saw operator and assistant

24. Question

Based on the concentrations of the radionuclides given in Table 5.2-2, the dose rates from the SFP water indicated in Section 5.2.2.3 appear too high. Notably missing from this table are ^{134}Cs, ^{137}Cs, and ^{60}Co concentrations which normally provide the major fraction of dose rate from the SFP water. Provide the data for these radionuclides for this table. Also, provide the concentrations of ^{134}Cs, ^{137}Cs, ^{58}Co and ^{60}Co following a refueling when the SFP water clean-up system is in equilibrium with the SFP water. Compare the calculated dose rates based on the total concentration in the pool with the 5 to 20 mrem/hr at the perimeter and pool center given in Section 5.2.2.3.

Response

Table 5.2-2(b) is an isotopic analysis of a sample of water drawn from Unit 1 and 2 Spent Fuel Pool March 19, 1979. Table 5.2-2(c) shows concentrations of Cs-134/137 and Co-58/60 present in Unit 1 and 2 Spent Fuel Pool following the refueling of Unit 1 when the pool water cleanup system was in equilibrium with the pool water (October 30, 1978). Based on the concentrations in Table 5.2-2(b), the dose rate calculated as being attributable to waterborne radioactivity within the pool is approximately 1 mr/hr. Figure 5.1 shows survey instrument readings made in the Unit 1 and 2 Spent Fuel Pool area on March 20, 1979.

No definite explanation for the apparent discrepancy in calculated and actual dose rates has been identified. The following effects are postulated as being contributors to such a disperity.

(a) Low dose rate error bands may be rather broad. The instrumentation is generally calibrated over a wide range with the readings experienced at the lower end of that range.

(b) Surface tension or other effects may cause the water activity to be stratified to such an extent that the actual water chemistry of the upper few mils of the pool water may be substantially higher than that of the sampled regions.

(c) Crud buildup along the walls due to wave-like actions of the water may be more substantial than anticipated.

TABLE 5.2-2(b)

GAMMA ISOTOPIC ANALYSIS OF UNIT 1 AND 2 SFP WATER

MARCH 19, 1979

Isotope	Concentration μCi/ml
Xe-133	7.6E-4
Xe-135	1.8E-4
Mn-54	1.2E-5
Co-58	1.4E-3
Co-60	6.9E-5
I-131	1.0E-4
I-133	1.1E-4
Ag-110m	4.7E-5
Cs-134	2.2E-3
Cs-137	3.4E-3

TABLE 5.2-2(c)

GAMMA ISOTOPIC ANALYSIS OF UNIT 1 AND 2 SFP WATER

OCTOBER 30, 1978
(AFTER REFUELING)

Isotope	Concentration μCi/ml
Co-58	2.7E-2
Co-60	4.2E-4
Cs-134	1.8E-3
Cs-137	2.7E-3

25. Question

Justify that the exposure received by the method that will be used to dispose of the present racks (i.e., cutting and packaging), as compared to crating the intact racks for disposal, will provide as low as is reasonably achievable (ALARA) exposure to personnel. Provide your considerations of costs, disposal volume and estimated exposure received for your proposed method and the alternative disposal methods. The estimated exposure should include the estimated dose rates, the number of people involved and their occupancy times. Explain in your discussion why the "fast cutting" torch to cut up the racks will only be used to minimize exposure above some minimum dose levels.

Response

The old racks will be removed in sections measuring approximately 5' x 7' x 14'. If they were to be boxed for LSA shipment and burial in these dimensions, burial costs alone would be approximately $50,700 (13,950 ft.^{3}; 28 boxes). The estimated exposure to accomplish this is 0.5 man-rem. The proposed method of reducing the volume of this low level waste is to cut the rack sections up in the fuel receiving area using a power saw. This should reduce the burial volume to approximately 720 ft.^{3}, on which burial costs would be approximately $3,500. The exposure estimate for this method is 2.0 man-rem. In burial costs alone, this represents a savings of over $47,000 for an additional exposure expenditure of 1.5 man-rem. It also conserves limited burial space. Localized high efficiency filtered exhaust ventilation system and contamination tents will be used, as necessary, for this cut-up and disposal method to ensure that exposures to personnel are ALARA.

The option of using the fast-cutting plasma arc torch is a contingency plan in the event that dose rates from the rack sections are very high and LSA shipment is not practicable. This method is described in Section 5.1.5 of the February 2, 1979 submittal. At low dose rates, this method could generate airborne activity problems, as well as smoke, metal fumes and gross levels of contamination. Thus, the internal exposure problems may equal or exceed the external exposure savings of this method if dose rates from rack sections are relatively low.

26. Question

Discuss in some detail the impact of the proposed SFP modification on radioactive liquid effluents from the plant, including leakage of water from the pool. Discuss the spent fuel pool leak collection system, including the disposition of leakage if it should occur, and the magnitude of leakage from the pool in the past.

Response

The spent fuel pool water has its own self-contained purification system. Under normal circumstances the spent fuel pool water has no direct input to the station liquid radwaste system. The only mixing of water occurs via mixing of reactor coolant with the pool water in the transfer canal. Any water from the pool would be reactor grade water and would have a negligible impact on either the reactor coolant or the liquid radwaste stream.

The spent fuel pool consists of a concrete structure with a leak tight stainless steel clad, steel plate liner. There is a leak collection system consisting of stainless channels inbedded in the concrete structure which supports the stainless steel clad liner. Any leakage from the liner would flow from these channels to one of the waste storage tanks. Enroute to the storage tank there is an open basin where the flow could be observed coming from the various pipes through open valves in this basin. This basin is inspected periodically for signs of pool leakage. From there the flow would drain into a common header leading to the waste tank. There has never been any leakage from the spent fuel pool and any that could occur during the modification of the pool would be detected through an increase in make-up water to the pool or an unusual increase in the level in the waste tank. Thus there should be no impact on the liquid waste system during or after the installation of the high density racks.

27. <u>Question</u>

Provide the average failed fuel fraction for the station since operation began.

<u>Response</u>

The station average failed fuel rate has been calculated to be 0.028% based of I-131 activity in the RCS.

28. Question

Provide the volume of a solid waste shipment from the replaceable pool filter and the frequency of its replacement under normal pool operation.

Response

It is possible to place only one filter in a 55 gallon drum which is equivalent to 7.35 ft^3. Both filters are normally changed at the same time to limit the required number of entries. The average frequency of change-out is two times per year. This is equivalent to 4 - 55 gallon drums or 29.4 ft^3/year. If the amount of cleanup required for the spent fuel pool water increased linearly with the increase in capacity the number of ft^3 produced per year could double producing 58.8 ft^3 of solid waste per year.

29. Question

Discuss the capability of the SFP cooling system to keep the expected, not design, SFP bulk water temperature at or below the FSAR design of 125°F during normal refuelings until the modified pool is filled. If the bulk water temperature is expected to be above the FSAR design value, discuss when this will occur and for what period of time. Include in this discussion, the effect of these higher temperatures on releases of tritium and radioiodine from the pool.

Response

The Spent Fuel Pool Cooling System is capable of keeping expected spent fuel pool bulk water temperatures at or below the FSAR design of 125°F during normal refueling operations until the modified pool is filled. The present system has this capability for current pool capacity, the augmented system will have this capability for the modified pool capacity. The augmented system will be installed prior to exceeding an assembly loading greater than current capacity. (Section 3.2.2 of the Submittal.) There will be no additional releases of tritium or iodine since the temperature will not be raised.

30. <u>Question</u>

Discuss the instrumentation to indicate the SFP water temperature. Include the capability of the instrumentation to alarm and the location of the alarms.

<u>Response</u>

The Spent Fuel Pool Temperature is monitored by a Foxboro-632 indicator/transmitter. The pool temperature can be read at the monitor locally or on the Unit 1 Computer. The computer monitor will alarm if the pool temperature exceeds 150°F. This computer alarm is displayed and printed in the control room.

31. Question

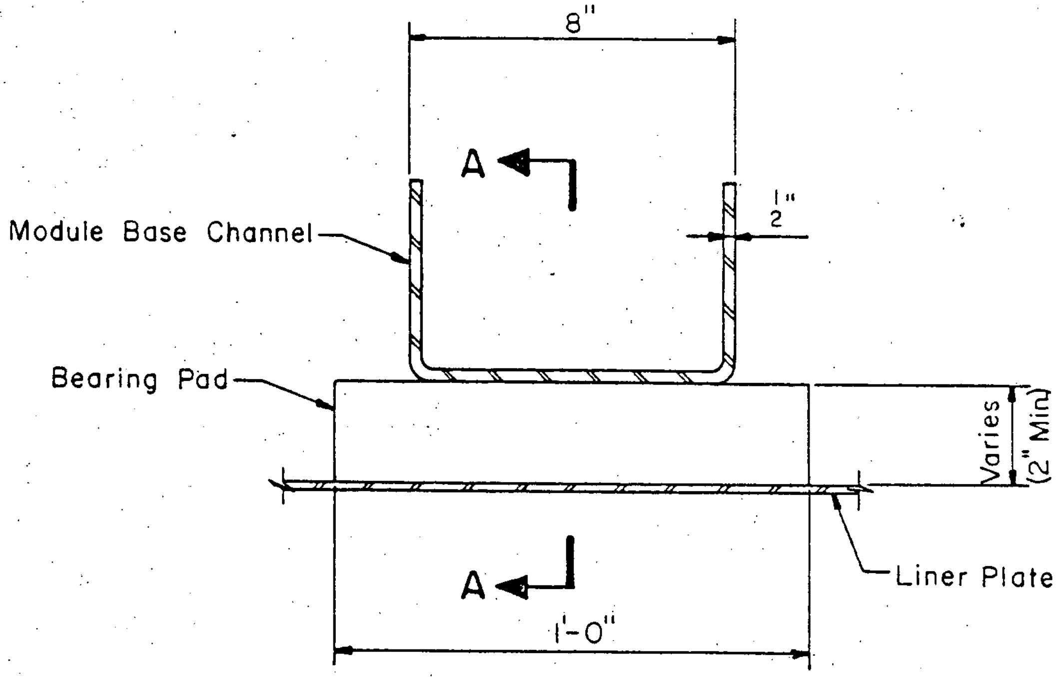
Propose a technical specification which prohibits carrying loads greater than the weight of a fuel assembly over spent fuel in the storage pool.

Response

A proposed charge to Oconee Nuclear Station Technical Specification 3.8.14 is included in Attachment 3.

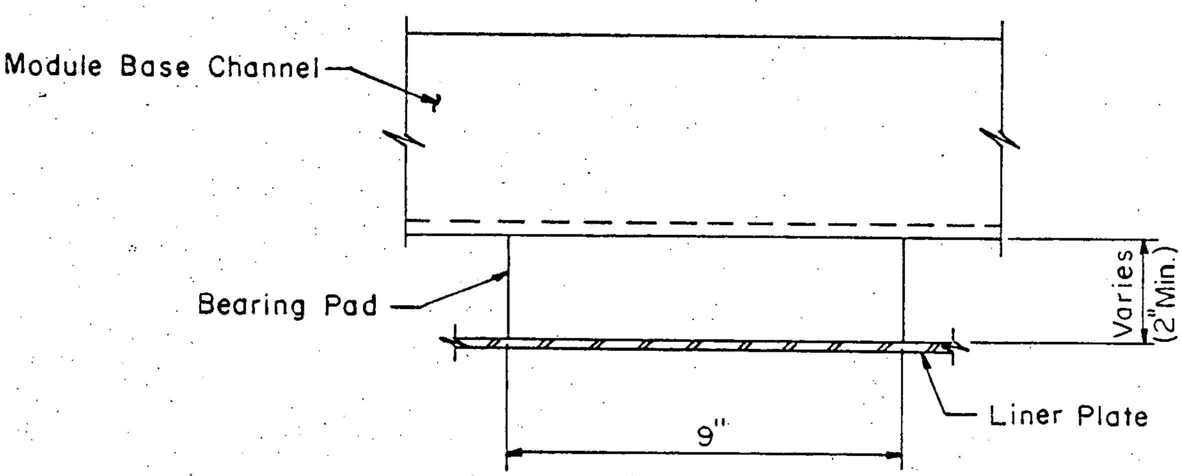
ATTACHMENT 2

NON-PROPRIETARY SKETCHES



TYPICAL BEARING PAD

Scale: None



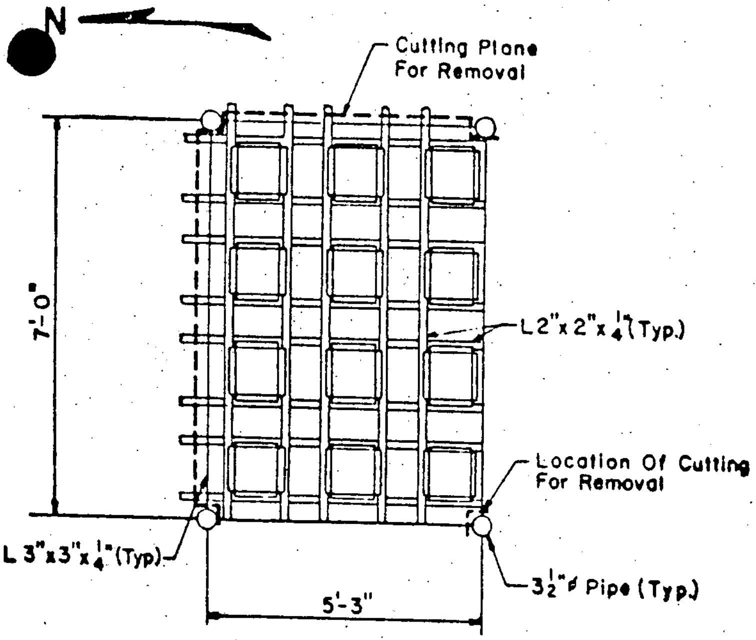
SECTION A-A

Scale: None

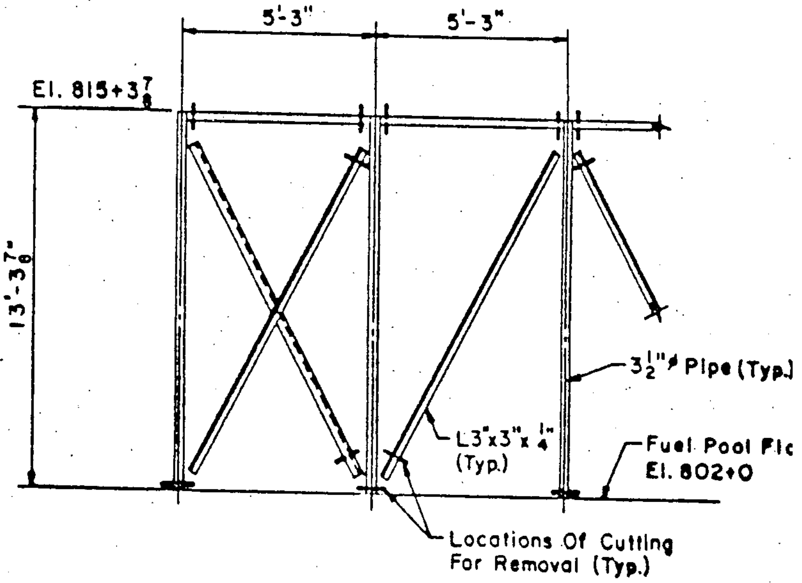
BEARING PAD SKETCH

OCONEE NUCLEAR STATION

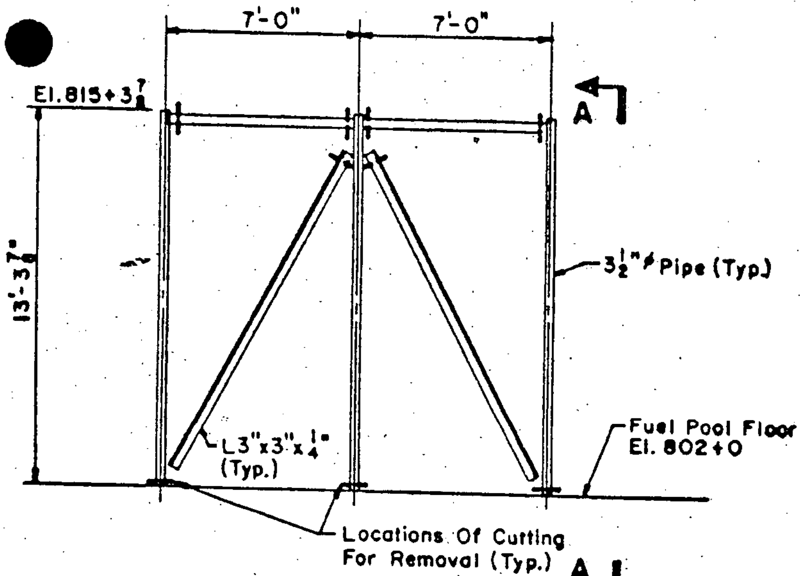
Figure 1



PLAN AT EL. 815+3 5/8
(TYPICAL BAY OF EXISTING RACKS)
Scale: None

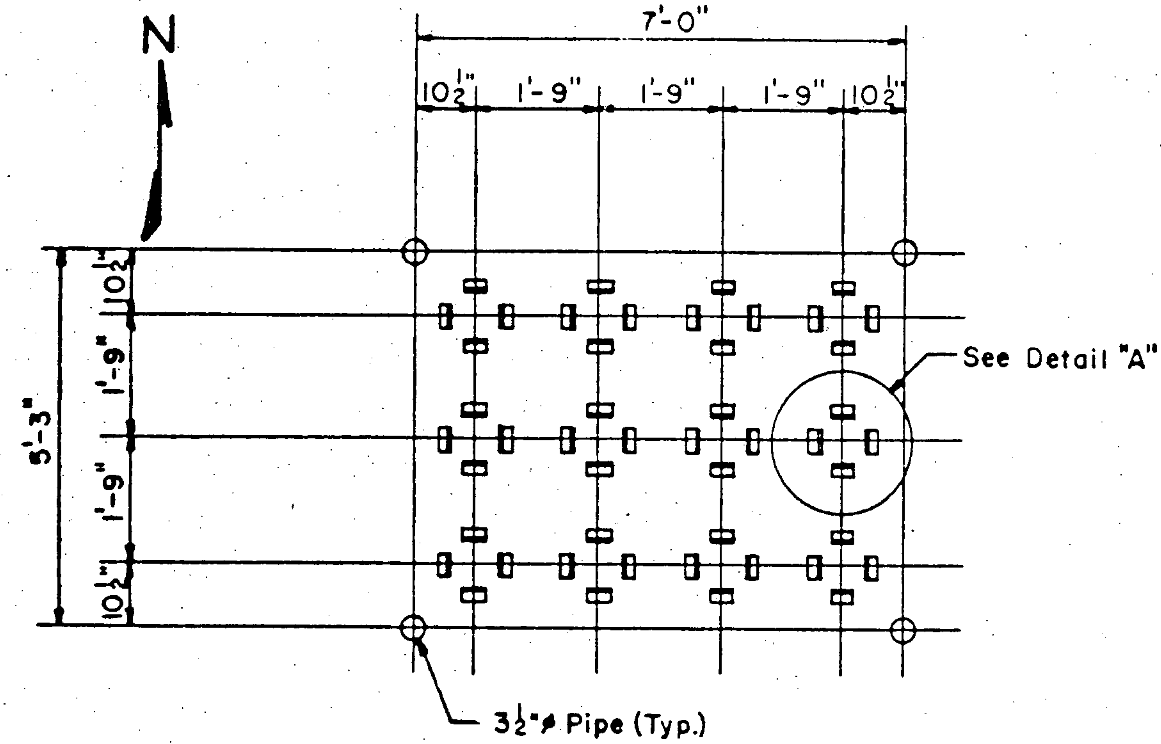


SECTION A-A
(TYPICAL NORTH-SOUTH BAY OF EXISTING RACKS)
Scale: None

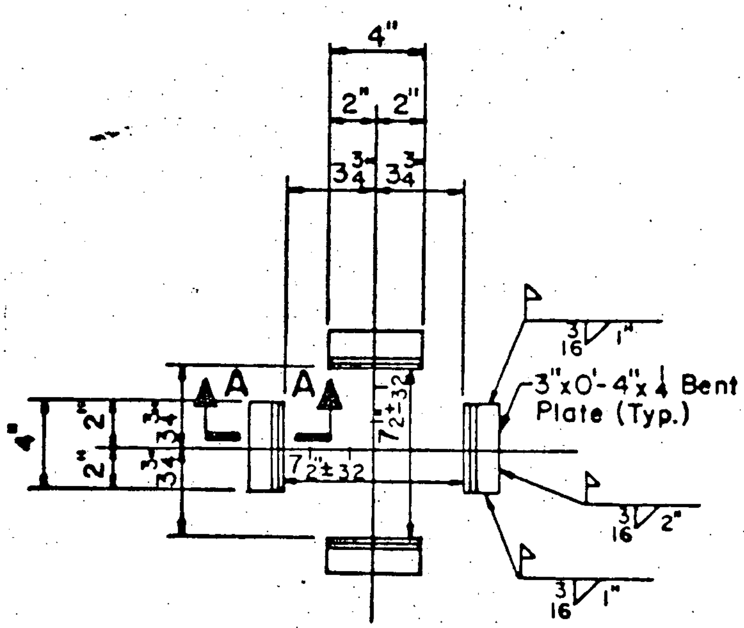


TYPICAL EAST-WEST
BAY OF EXISTING RACKS
Scale: None

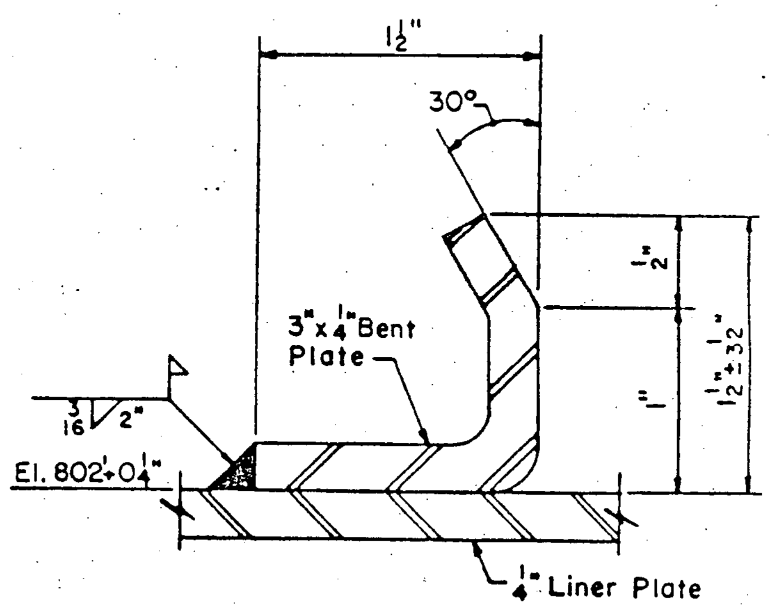
EXISTING RACK STRUCTURES
("PIPES" AND "ANGLES" SKETCH)
DUKE POWER
OCONEE NUCLEAR STATION
Figure 2



BOTTOM PLAN
TYPICAL BAY OF EXISTING RACKS
Scale: 3/8" = 1'-0"



DETAIL "A"
(TYPICAL FOR EACH ASSEMBLY LOCATION)
Scale: 1½" = 1'-0"



SECTION A-A
Full Scale

DUKE POWER CITIZENSHIP SERVICE

EXISTING RACK STRUCTURES
("BENT PLATES" SKETCH)
OCONEE NUCLEAR STATION
Figure 3

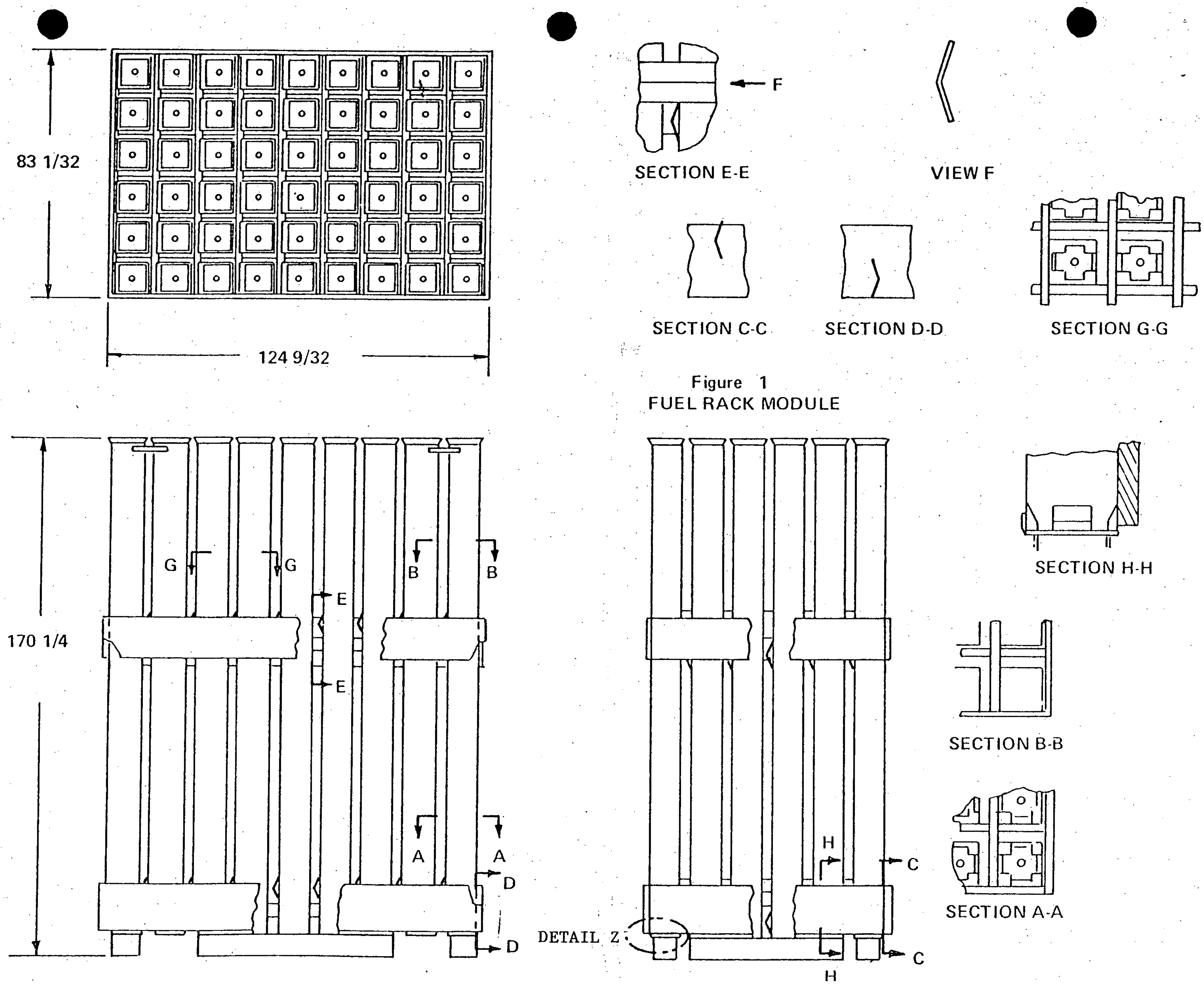
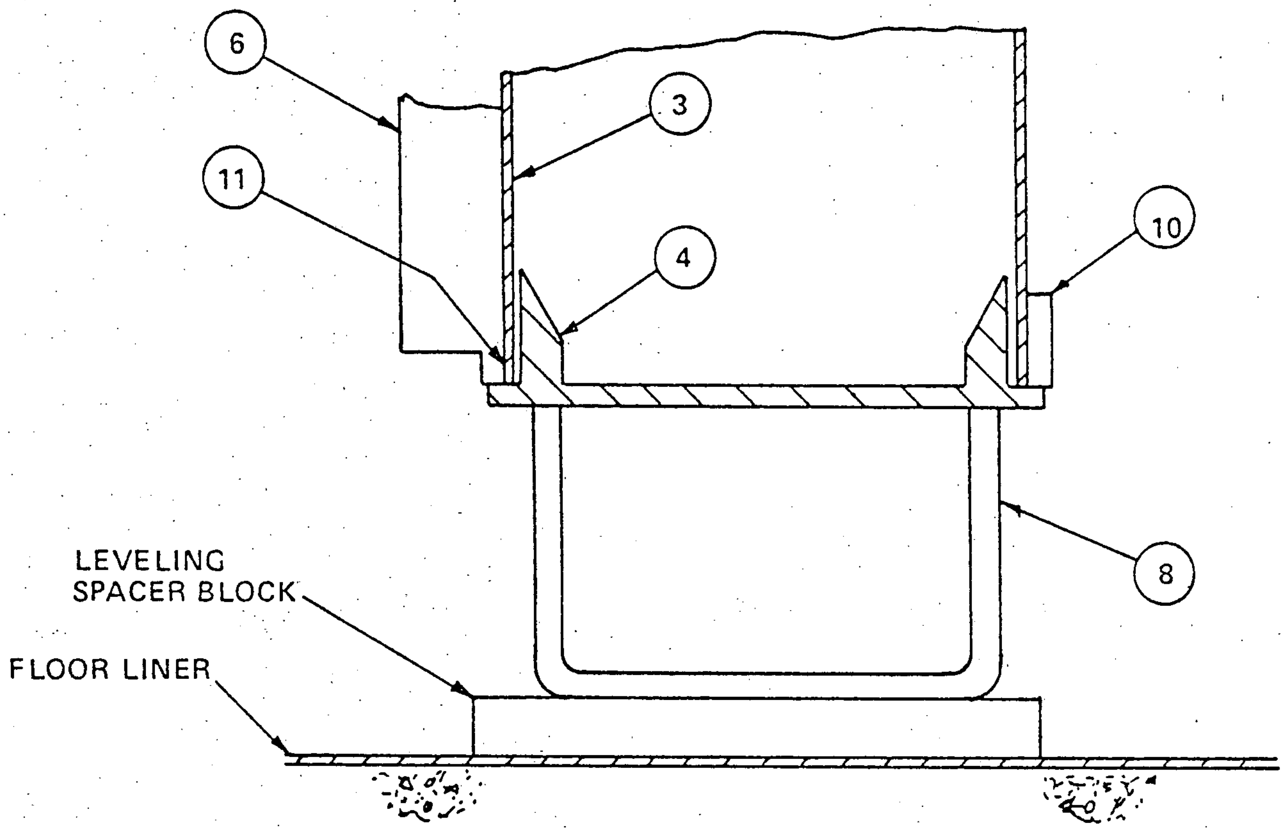


Figure 1
FUEL RACK MODULE

RACK MODULE DETAILS
OCONEE NUCLEAR STATION
Figure 4



3 CAVITY

4 CAVITY BOTTOM PIECE

6 PERIPHERAL WRAPPER PLATE

8 SUPPORT CHANNEL

10 11 TRANSITION PIECES

DETAIL Z OF FIGURE 4
OCONEE NUCLEAR STATION
Figure 5

ATTACHMENT 3

TECHNICAL SPECIFICATION CHANGE

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 moved 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 suspended loads of more than 3000 lb_m shall be transported over spent fuel stored in either spent fuel pool.

3.8.15 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 $k_{eff} = 0.99$ 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 doese for the fuel handling accident are within the guidelines of 10CFR100; 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 3.8.13 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.

Specification 3.8.14 is required to prohibit transport of loads greater than a fuel assembly with a control rod and the associated fuel handling tool(s).

REFERENCES

(1) FSAR, Section 9.7
(2) FSAR, Section 14.2.2.1
(3) FSAR, Section 14.2.2.1.2