Florida Power
CORPORATION

W. P. STEWART, DIRECTOR
POWER PRODUCTION

March 16, 1979

Mr. Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72

Dear Mr. Reid:

In our letter of January 18, 1979, Florida Power Corporation indicated its decision to delay installation of the high density spent fuel storage racks at CR #3 until after the first refueling outage. This decision was made primarily to allow time for resolution of the generic licensing problems regarding the use of a poison material manufactured by Carborundum Company.

Florida Power Corporation still wishes to pursue resolution of all NRC concerns related to our docket while the generic issue is being resolved. In this regard, we hereby submit, for your staff's review, three (3) originals and forty (40) copies of our response to your request for additional information dated August 1, 1978. This submittal completes our response to all outstanding questions of your staff.

Florida Power Corporation stands ready to discuss any of our submittals with your staff in order that NRC approval of our modification can be achieved in a time frame consistent with resolution of the generic licensing issues.

Please contact this office if you or members of your staff require any additional information or discussion concerning our submittals.

Very truly yours,

FLORIDA POWER CORPORATION

W.P. Stewart

W.P. Stewart

File: 3-0-3-a-3

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STATE OF FLORIDA

COUNTY OF PINELLAS

W.P. Stewart states that he is the Director, Power Production, of Florida Power Corporation; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.

W.P. Stewart

W.P. Stewart

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 16th day of March, 1979.

Charles R Pope

Notary Public

Notary Public, State of Florida at Large,
My Commission Expires: July 25, 1980
(Notary 1 D12)

RESPONSE TO NRC REQUEST

FOR ADDITIONAL INFORMATION

HIGH DENSITY RACK MODIFICATION

CRYSTAL RIVER UNIT 3

March, 1979

ENCLOSURE 1

Question 1

Discuss the occupational exposure expected during the SFP modification including the prepartory work and considering the pool is contaminated. Address the expected dose rates, numbers of workers (including divers, if necessary) and occupancy times for each phase of the operation. Include removal and disassembly (or crating) and disposal operations of the low density spent fuel racks and installation of the new high density racks. Provide the resultant man-rem exposure.

Response 1

The SFP modification can be divided into four phases of operation which are:

Phase I - Decontamination of Pool A and Preliminary Work
Phase II - Rack Removal
Phase III - Rack Disposal
Phase IV - Installation of New High Density Racks and Clean-up Activities

The total manpower estimate (craft labor) for the entire rack modification is approximately 12,530 man-hours. The number of workers that will be utilized will be dependent upon the actual exposure rates at the time of the modification.

For Phase I, the spent fuel pool will be decontaminated using hydro-lasers followed by a vacuum and filtration of the final 6" of water in the bottom of the pool. The average dose rate in the pool during this phase is expected to be 25 mr/hr. The SFP modification will be performed with Pool A drained, therefore, no divers will be required.

For Phases II and III, the expected dose rate following decontamination in the pool will be 15-25 mr/hr.

For Phase IV, following rack removal, the dose rate in the spent fuel pool should be less than 15 mr/hr.

During the modification of the racks at CR #3, the man-rem exposure to the workers will be limited to less than 300 mrem/man/week. If written approval from the Chem-Rad Section is obtained, this limit may be raised to 600 mrem/man/week.

We are presently performing work on the fuel transfer mechanisms located in Pool A at CR #3. This working situation is very similar to the conditions expected during the upcoming rack modifications, in that Pool A has been drained, decontaminated and spent fuel is presently being stored in Pool B. The personnel exposures for this work effort are averaging around 150 mrem/man/week.

Question 2

If the low density racks are to be cut up for disposal, explain why the exposures received by personnel would be as low as reasonable achievable (ALARA) as compared to crating the low density racks intact.

Response 2

FPC is presently exploring the feasibility of shipping the present spent fuel storage racks from Pool A offsite intact as well as cut up into pieces. Once a definite decision has been reached based on personnel exposure, cost/benefit, and feasibility, FPC will supply additional information justifying our decision.

Question 3

Identify the principal radionuclides and their respective concentration in the spent fuel pool as a result of placing the core in the pool during steam generator repair operations.

Response 3

The activity levels of the Reactor Coolant System have been maintained between 5×10^{-5} to 1×10^{-4} uci/ml. The principal radionuclides and their respective percentage of concentration are: Co^{58}-85%, Co^{60}-5%, Mn^{54}-3%, Cr^{51}-3%, Co^{137}-2%, Nb^{95}-1%, others -1%.

Question 4

Provide the dose rates above and around the spent fuel pool from the concentrations of the radionuclides identified in 3 above. Also, provide the estimated dose rate of the contaminated racks when they are removed from the spent fuel pool.

Response 4

Dose rate above and around pool, Full	- < 1 mr/hr
Dose rate above and around pool, Drained before decontamination	- < 5 mr/hr
Dose rate above and around pool, Drained after decontamination	- < 1 mr/hr
Dose rate of racks after removal	- ~10 mr/hr

Question 5

Discuss the capability of the Spent Fuel Pool Cooling System to keep the actual spent fuel pool bulk water temperature at or below the FSAR design of 120° 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. Discuss

also the impact of any expected higher than design value pool temperatures on the gaseous releases of radioiodines and tritium from the pool.

Response 5

The maximum bulk water temperature under normal operating conditions with the modified pool fully loaded is 125°F. The spent fuel coolers are designed for temperatures well in excess of the above referenced 125°F. Although the proposed bulk water temperature is in excess of the FSAR value, the 5°F increase is considered to be inconsequential relative to the overall design and operating conditions of the SFSP cooling system.

The impact of the 5°F increase in the bulk water temperature is considered to be no more significant than the daily fluctuation of the atmospheric conditions and is negligible in regards to total plant gaseous effluent.

Question 6

Provide the estimated volume of contaminated material (e.g., spent fuel racks, seismic restraints) expected to be removed from the spent fuel pools during the modification and shipped from the plant to a licensed burial site.

Response 6

The estimated volume of contaminated material for shipment to a licensed burial site or vendor is:

1) Four fuel racks - 7' x 9' x 14' each
2) Two fuel racks - 7' x 11' x 14' each

If these racks are decontaminated and cut up for disposal prior to shipment from CR#3, the estimated volume could be reduced approximately 50%. This is based on cutting the racks into 4' sections and packing them for shipment.

Question 7

Provide a list of typical loads that might be carried near or over the spent fuel pool. Provide the weight and dimensions of each load. Discuss the load transfer path, including whether the load must be carried over the pool, the maximum height at which it could be carried and the expected height during transfer. Provide a description of any written procedures instructing crane operators about loads to be carried near the pool. Provide the number of spent fuel assemblies that could be damaged by dropping and/or tipping each typical load carried over the pool.

Response 7

The attached Table I and Figure I identify the loads that might be or are carried over the spent fuel pool. The following is a list of the plant

procedures which address the movement of loads near the pool. Copies of these procedures were furnished to the NRC in our submittal of August 18, 1978 concerning movement of heavy loads near spent fuel.

1) SP-434, Fuel Storage Pool Missile Shields
2) SP-530, Demonstration of the Auxiliary Bldg. Overhead Crane (FHCR-5) Interlock Operability
3) SP-531, Spent Fuel Bridge Interlock Surveillance
4) SP-532, R.B. Main & Auxiliary Bridge Interlocks
5) SP-601, Procedure for Load Testing Sling and Lifting Fixtures
6) SP-671, Spent Fuel Bridge Load Test
7) SP-672, Procedure for Load Testing New Fuel Elevator
8) FP-203, Defueling and Refueling Operations
9) FP-601, Fuel Handling Equipment Operations
10) FP-1001, Spent Fuel Handling

The utilization of a load list is of no specific merit. The fact that a load size, dimension, weight or configuration is known does not provide sufficient information in regards to the number and extent of fuel assemblies that could be damaged. It is essentially impossible to predict a load's angle of attack and final resting position for each of the above listed loads. However, Technical Specification 3.9.7 requires that loads in excess of 2,750 pounds, except for movement of the missile shield and pool divider gate as necessary for access to the fuel assemblies, be prohibited from travel over fuel assemblies in the storage pool. The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses. The radiological consequences due to the rupture of 208 pins (that of a spent fuel element) at maximum activity has been analyzed and determined to be small and is referenced in the "Safety Evaluation Report."

Question 8

Discuss the instrumentation to indicate the spent fuel pool water level and water temperature. Include the capability of the instrumentation to alarm and location of the alarms.

Response 8

CR #3 has monitoring equipment which will alarm if the fuel pool water level falls below or rises above a predetermined level. The water level alarm will annunciate in the control room if the water level drops 4 feet below or rises 3 feet above the reference water level (Elev. 158'-6"). The low water level alarm is set above the minimum technical specification requirement for maintaining at least 23 feet of water over the top of irradiated fuel assemblies.

CR #3 does not have any alarm instrumentation to direct measure and alarm the spent fuel water temperature in the control room. However, the spent fuel cooling system as discussed in response #5 is designed to maintain

the fully loaded spent fuel pool at an acceptable temperature. In the event that a spent fuel coolant pump trips the operator will receive an alarm in the control room. In addition, local temperature readings are taken at the spent fuel coolers by the operators in accordance with CR #3 surveillance procedures. Therefore, by verifying proper operability of the spent fuel cooling system we indirectly monitor the temperature of the spent fuel pool water.

Question 9

Your March 3 and 22, 1978 submittals did not address the impact of the proposed SFP modifications on the environmenta. Discuss in some detail the impact of the proposed SFP modification on the following:

a. radioactive gaseous effluents from the pool, and

b. radioactive liquid effluents from the plant, including leakage of water from the pool and the SFP leak collection system.

Response 9

a. As discussed in the response to Enclosure 1, item 5, the impact of the proposed modification on gaseous effluents is considered to be insignificant. The impact on the environment is considered to be negligible because the contribution of the SFP on plant gaseous effluents is much less than that resulting from normal leakage in the auxiliary building.

b. The internal modification of the spent fuel storage racks within the pool has no impact on the radioactive liquid effluent values as stated in the FSAR. The radioactivity in the pool water is introduced during refueling and is subsequently removed by the cleanup system. Presence of fuel assemblies previously moved to the pool will not affect the liquid activity levels.

Question 10

Your March 3 and 22, 1978 submittal did not propose changes to the SFP purification system. Discuss in some detail why the present SFP purification is adedquate for the proposed SFP modification. Include the experience of operating the SFP with a full core in the pool during steam generator repair operations with the typical dose rates in the vicinity of the pool and the frequency of replacing the demineralizer resin and filters.

Response 10

The Spent Fuel Pool purification system is considered adequate for the following reasons:

a. The Spent Fuel Pool purification system is presently capable of handling the existing SFP volume which has not increased due to SFP modification to store additional Spent Fuel Assemblies.

b. The maximum load originally to be considred was a full core unload. This is also the basis with the SFP modification.

c. The long term storage of spent fuel in 1/3 (one-third) core increments is not an apprciable impact for original SFP purfication system or under SFP modification to store additional Spent Fuel Assemblies.

During previous outages the typical dose rate in the vicinity of the spent fuel pools was less than 1.0 mr/hr. During a one month period it was necessary to change out filters at a rate of one per day and to change the resin bed once during the 30 day period. The dose rate at contact of the filters was approximately 150-500 mr/hr and the resin bed dose rate at contact was < 5R/hr. This excessive replacement of filters and resin during this 30 day period was not due to the spent fuel in the pool but due to an external source of free carbon steel (shipping liners introduced to the boric acid). Once the liners were removed and the water chemistry was returned to normal specifications, the filters were changed out at a rate of once per quarter and the resin bed was changed once per six months with spent fuel in the pool.

Question 11

At present there are four spent fuel assemblies in the pool which might remain in the pool during the modification. Discuss what effects these four assemblies have on implementing the pool modification including exposure to workers and the possibility of dropping heavy loads on spent fuel.

Response 11

As a result of the first refueling outage at CR #3, there will be 60 spent fuel assemblies located in Pool B during the modification of Pool A. Exposure to workers will be minimal because the fuel in Pool B will be isolated from Pool A through the use of shielding (walls, pool gate, water, missile shields, etc.) and distance. The missile shields will prevent damage to any of the spent fuel assemblies in the unlikely event a load is dropped over Pool B.

TABLE I

LIST OF LOADS MOVED OVER SPENT FUEL STORAGE POOLS

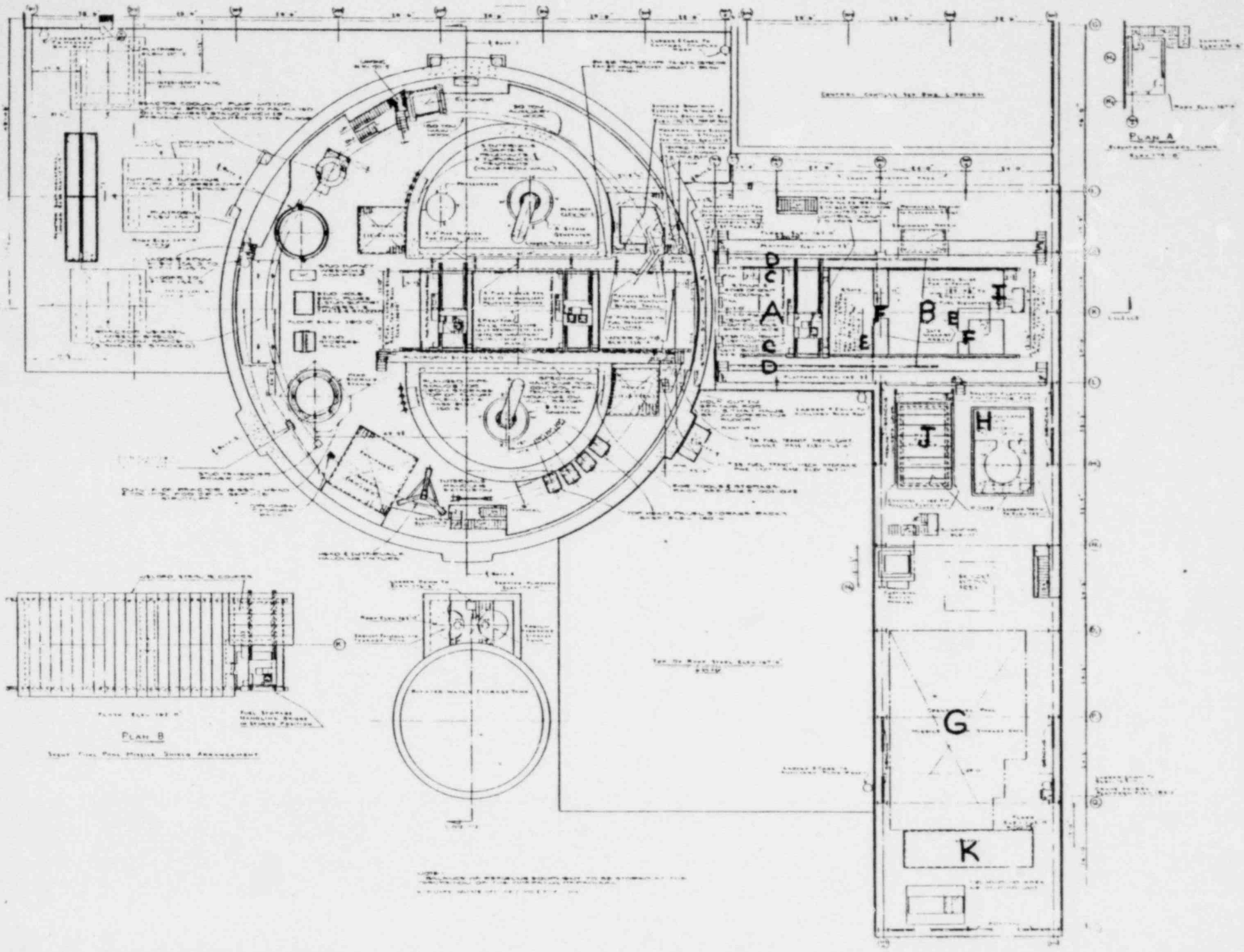
OBJECT	WEIGHT (lbs)	SIZE	MOVEMENT PATH (1)	HEIGHT (ft)(2)	FREQUENCY OF MOVEMENT
Fuel Assembly	~ 1550	Length - 14' Width - 8.5" Depth - 8.5"	A to B J to I to C to Reactor (New Fuel)	4	Refueling
Transfer Carriages	~ 2626	Length - 24' Width - 2'	C to D	30	Only for maintenance or repair
Lifting Beam for Carriages	~ 300	Length - 8' Width - 8"		33	Used for carriage removal
Spent Fuel Gates	~ 3900	Length - 28' Width - 35.5" Thickness - 1"	E to F	30	Refueling - moving fuel between pools
Missile Shields	~ 8200	Length - 26' Width - 30" Height - 30"	G to B to A	37	Refueling, maintenance
New Fuel Elevator Test Weight	~ 2080	Length - 16' Diameter- 8"	H to I	37	During new fuel receipt
Equipment/Personnel Lifting Basket	~ 250	Length - 4' Width - 4' Height - 3'	Throughout Fuel Pool Area	30	During maintenance on pools or equipment

NOTES TO TABLE I

(1) The movement path locations designated in this table are shown on Figure 2 attached.

(2) The heights indicated are approximate maximum heights above the top of the spent fuel storage racks.

Figure I



ENCLOSURE 2

A. In Florida Power Corporation's March 3, 1978 submittal, it is stated that Carborundum Company's B_4C Composite material was selected for the Crystal River storage racks and that the Carborundum Company has initiated a qualification test program similar to that performed for the B_4C Plates which were used in several other storage racks such as those for the Haddam Neck plant. However, due to the recent experience with those racks, this qualification program needs to be reexamined in detail. The following information is needed for this reexamination:

Question 1

Will any sources of neutrons other than spent fuel assemblies be stored in the spent fuel pool? If so, at what rate will they emit neutrons?

Response 1

No sources of neutrons other than spent fuel assemblies will be stored in the Crystal River spent fuel pools. It is not anticipated that neutron sources will even be used for any testing within the spent fuel pool.

Question 2

What is the melting temperature of the boron containing material in the unirradiated condition?

Response 2

The B_4C Composite material can operate continously at temperatures up to 350°F. While the materials within the Composite system do not melt, the most temperature sensitive component begins converting to carbon at 450°F.

During residence in the spent fuel pool, the maximum temperature at which the Composite material will operate is well below 350°F (see Response 4). During fabrication, the fuel storage cell will be assembled by welding with the Composite material in place. Consequently, the storage cell has been designed to preclude the Composite material from reaching an excessive temperature during fabrication (more than 450°F). This is accomplished by 1) providing a suitable distance between the weld regions and the boron containing material, 2) providing significant heat sinks in the vicinity of the welds (e.g. storage cell angles) and 3) selecting weld processes which minimize the heat input into the weld region (e.q. resistance weld).

Question 3.

What will the maximum integrated neutron and gamma flux be in the boron containing material over the lifetime of the racks? What spent fuel assembly power density and burnup, and what rack life were assumed in calculating these maximum integrated fluxes? What is the assumed energy spectrum for the gamma flux?

Response 3

An analysis was performed to establish the maximum integrated neutron and gamma dose to which the B_4C Composite material would be subjected over the lifetime of the racks (40 years). The calculations were performed using the ANISN computer program which modeled an infinite array of Crystal River fuel assemblies using cylindrical geometry. The following parameters were assumed for the fuel assembly array:

Fuel assembly power density (average for 4 years)	12.1 MW/assembly
Fuel assembly burnup (for 4 years)	38,000 MWD/MTU
Gamma spectrum	Used average gamma energy of 0.8 MEV
Integrated exposure time	From 1 second up to 40 years

The results of the analysis indicate that the contribution of neutron irradiation to the total dose received by the Composite material is negligible compared with the dose resulting from gamma irradiation. The following gamma irradiation data summarizes the results of the analysis for the principal exposure times:

Dose at 1 year exposure	6.7×10^9 Rad
Time required to reach 10^{10} Rad	3 years
Dose at 40 years exposure	1.8×10^{10} Rad

The short-term phase of the Carborundum qualification test program has exposed the Composite material to 10^{10} Rad gamma with the results showing that the material retains acceptable properties (see Response A6). The long-term phase is scheduled to expose the material to 10^{11} Rad gamma. As can be seen, the results of the short-term program cover a major portion of the exposure anticipated for the life of the fuel racks and the long-term program irradiation will clearly exceed the anticipated life exposure (10^{11} Rad vs. 1.8×10^{10} Rad). The results of the long-term program (scheduled for completion in February, 1979) will be available well before the Crystal River racks store any discharged fuel (installation is scheduled for mid-1979). The listed

exposure values are predicated on the reasonable assumption that a fuel assembly placed in a storage cell will remain in that position until it is shipped from the site. A more conservative exposure basis is to assume that fuel assemblies will be removed from storage locations every five (5) years during the 40 year life of the rack and replaced with freshly discharged assemblies. This 5 year period is consistent with the cooling times that are being mentioned as prerequisites for shipment to Away From Reactor (AFR) Storage Facilities. Using the assumption that a freshly discharged fuel assembly is placed in every storage location every 5 years, the resulting total exposure of the Composite material is calculated to be 9.6×10^{10} Rad for the 40 year rack life. This value is consistent with the total exposure scheduled for the long-term phase of the Carborundum qualification test program (10^{11} Rad).

The most conservative exposure basis for any boron containing material is to assume that spent fuel assemblies will be removed from the storage locations every refueling outage and replaced with freshly discharged assemblies. This exposure basis can occur consistent only if the current Crystal River spent fuel handling procedures are modified to force the yearly replacement. A spent fuel handling procedure which tends to maximize fuel handling will not be permitted by FPC because it is contrary to FPC's objectives to 1) reduce the possibility of fuel damage during handling and, therefore, the possible release of radioactive material to the pool environment and 2) reduce the exposure of plant personnel to the radiation environment existing at the surface of the spent fuel pool during fuel handling. Consequently, FPC concludes that the application of this exposure basis to the Composite material is not justifiable. Even if it is assumed that an accidental replacement of spent fuel will occur on a yearly basis, it would take about 15 years before the Composite material reached an exposure limit of 10^{11} Rads. Clearly, ample opportunity would exist to terminate repetitive accidental replacements before the 10^{11} Rad exposure limit was reached.

Question 4

What will the maximum temperature be in the center of the boron material, assuming the highest neutron and gamma flux and the worst accident conditions?

Response 4

The interior temperature of the B_4C Composite is determined by the temperature rise in the material due to gamma density (the neutron flux levels are insignificant and produce negligible heating) and the temperature of the spent fuel pool water which is in contact with the storage cell walls.

The maximum temperature rise in the Composite material occurs when freshly discharged fuel is placed in the storage cell (and the adjacent storage cells). The maximum dose rate associated with freshly discharged fuel has been calculated to be 2×10^6 Rad/hr. using the calculational model described in Response 3 and assuming 5 days of cooling. The temperature rise for this dose rate will be less than 2°F. This value is based on temperature test data obtained in the Carborundum qualification test program. (see figure on page E-1 of Reference 1).

The 2°F temperature is conservative for Crystal River since the test data was obtained for the relatively thick B_4C Plate material (0.210 inches thick) placed dry in the stainless steel sample holders and in casual contact with the bottle walls while the thinner Composite material in the Crystal River racks (~0.05 inches thick) is immersed in the pool water and is in intimate contact with the storage cell walls.

The maximum temperature of spent fuel pool water in contact with the storage cell walls is primarily a function of the bulk water temperature of the spent fuel pool. Under the worst heat load conditions (full core discharge and single failure of the pool cooling system), the Crystal River spent fuel pool may have a bulk water temperature of 205°F. Thermal hydraulic analyses performed for the Crystal River fuel racks indicate a maximum temperature rise (within the storage cell) of 28.0°F. Consequently the maximum temperature of pool water contacting the storage cell walls is 233.1°F.

Combining the maximum temperature rise (2°F) and the maximum pool water temperature (233.1°F) results in a maximum temperature of ~235°F for the Composite material. This temperature is well below the steady state operating temperature of the Composite material (350°F).

Question 5

What will the chemical composition of the boron containing material be after receiving the design dose or irradiation?

Answer 5

The chemical composition of the B_4C Composite material after exposure to 10^{10} Rad gamma is analytically the same as the chemical composition prior to irradiation with an estimated 0.2 w/o loss of hydrogen. It is expected that the chemical composition after 10^{11} Rad exposure will also be essentially the same with, however, a proportionately larger loss of hydrogen.

Question 6

What will the physical properties such as the density, the modulus of rupture, the modulus of elasticity, and the compressive strength of the boron containing material be after it receives the design dose or irradiation in the spent fuel pool?

Response 6

A principal objective of the qualification test program initiated by the Carborundum Company is to establish the mechanical and physical properties of the B4C Composite material proposed for the Crystal River racks as a function of the gamma irradiation dose with simultaneous exposure to prototypical pool water environments. At this point, the mechanical* and physical properties have been evaluated for gamma irradiation doses up to 10^{10} Rads with simultaneous immersion in demineralized (D.I.) water and in borated water (2500 PPM). In addition the properties have been determined for: 1) exposure to 10^{10} Rads only and 2) exposure to demineralized or borated water only. A test to establish the mechanical and physical properties for exposure to 10^{11} Rads with simultaneous immersion in demineralized water has been initiated by Carborundum and is scheduled to be completed by mid-February. As indicated in Response 3, the available test data covers a major portion of the gamma exposure anticipated for the life of the Crystal River fuel racks (10^{10} Rads vs. 1.8×10^{10} Rads) and that the data being developed for 10^{11} Rads will bound exposure levels associated with the fuel stored in each storage location periodically replaced by freshly discharged fuel at a frequency in excess of any operationally justifiable value.

Table 1 presents the summary results of the qualification test program to date. Specifically the table presents the principal mechanical and physical perperties evaluated during the test program and the changes in those properties that resulted from combinations of exposure to gamma radiation and prototypical pool water environments.

The following observations and conclusions can be made from Table 1:

a. The changes in mechanical properties are more pronounced for the Composite material immersed in D.I. water than in borated water. Consequently the long term testing to 10^{11} Rads is being carried out in D.I. water.

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* The mechanical properties of interest for the Composite material are the Ultimate Tensile Strength (UTS) and the Modulus of Elasticity (MOE). The Modulus of Rupture (DOR) does not apply to the Composite material.

b. The change in the principal mechanical properties (UTS) of the Composite material immersed in D.I./borated water only are similar to the changes observed for the material immersed in D.I./borated water and simultaneously exposed to 10^{10} Rads.

c. The mechanical properties of dry Composite material irradiated to 10^{10} Rad are essentially the same as the properties of the unirradiated baseline material.

d. Considering items b and c it is evident that up to 10^{10} Rad, the decrease in mechanical property values for the composite material simultaneously exposed to radiation and water is due primarily to immersion of the material in either the D.I. or borated water. With respect to water immersion, the tests indicate that essentially all of the observed change in mechanical properties due to immersion occurs over the first 72 hours of immersion. After that time, the rate at which additional property changes occur becomes negligible so that no additional significant changes in the mechanical or physical properties due to immersion in the pool water are expected over the 40 year life of the Crystal River fuel racks.

e. The changes in the physical properties of the Composite material are negligible up to 10^{10} Rads.

f. The changes in the mechanical properties are modest up to 10^{10} Rad with simultaneous immersion in either D.I. or borated water (25% or less of the baseline values). The mechanical property values remain well above the requirements specified in the Composite material purchase specification. The minimum specified UTS (Ultimate Tensile Strength) value (2200 psi) is significantly greater than the maximum stresses that the Composite material will experience during a SSE event (52 psi). The calculated stress value includes the effect that fuel assembly rattling has on the displacement of the fuel rack storage cell during the SSE.

g. The predicted UTS value for 10^{11} Rad with simultaneous exposure to D.I. water (4500 psi) is also significantly greater than the minimum specified UTS value (2200 psi) and the maximum tensile stress that the Composite material will experience. The predicted value was conservatively developed from a comparison of the 10^{9} 10^{10} Rad gamma radiation test data with data obtained from electron beam irradiation tests (up to 2×10^{11} Rad) performed previously for the Composite materials. Compressive strength values

TABLE 1

COMPOSITE MATERIAL TEST DATA

GAMMA EXPOSURE (Rad)	BASELINE DATA	WATER IMMERSION		SHORT TERM TEST PROGRAM DATA					PROJECTED EXPOSURE DATA	MINIMUM SPECIFICATION VALUE	MAXIMUM CALCULATED STRESS VALUE
	0	0	0	10^9 RADS		10^{10} RADS			10^{11}	10^{11}	
WATER EXPOSURE	DRY	DI	BORATED	DI	BORATED	DRY	DI	BORATED	DI	DI OR BORATED	
Mechanical											
UTS (psi)	8838	7300	7969	8457	8160	9406	6591	7695	4500	~2200	52
% of Baseline	100	82.6	90	95.7	92.3	106.4	74.6	87.0	50.9	25	(SF = 2200/52 = 42)
MOE (psi x 10^{-5})	2.24	2.28	2.27	1.64	1.67	2.26	1.73	1.75			
% of Baseline	100	101.8	101.3	73.2	74.5	100.9	77.2	78.1			
Physical											
Length (m)				3.004	3.005		2.999	2.999			
% Change				-0.5	+0.2		0	+0.1			
Width (m)				0.538	0.531		0.533	0.522			
% Change				+0.9	+0.2		0	0			
Thickness (m)				0.048	0.048		0.049	0.048			
% Change				+2.1	0		4	0			
Weight (gm)				1.757	1.773		1.767	1.733			
% Change				-0.23	-0.2		-1.1	-0.3			

were not determined in the Carborundum test program for the Composite material. However, it is evident from the nature of the material and from compressive strength data available for the Plate material (a boron containing material which has the same binder as the Composite) that the compressive strength will be superior to the tensile strength (UTS values). Calculations for compressive loading of the Composite material under SSE conditions with fuel assembly impact show that the anticipated compressive stresses are extremely modest compared with the UTS values (~25 psi vs. 2200 psi minimum value permitted by the purchase specification).

Question 7

Provide data to show that, under the combined effects of irradiation and immersion in borated fuel pool water, the leachability of the boron will not be synergistically enhanced over the life of the high density storage racks.

Response 7

The rates at which boron is leached from B_4C Plate material (a material similar to the B_4C Composite proposed for Crystal River) were established for two environmental conditions: 1) immersion in demineralized water with no exposure to radiation and 2) immersion in demineralized water with simultaneous exposure to gamma radiation. Because of the nature of the test in which small concentrations of leached boron must be detected, it was not possible to immerse the boron containing material in borated water. Immersion in demineralized water, however, is conservative since borated pool water with its relatively high boron concentration will inhibit leaching.

The results of the leaching tests for the two environments are presented graphically on page H-5 of Reference 1. The test data indicate that the leaching behavior for the two environments are sufficiently similar to conclude that boron leachability is not synergistically enhanced by the presence of gamma radiation.

An analysis of the test data obtained with simultaneous irradiation indicates that approximately 0.23% of the available boron will be leached from the tested boron containing material over a 40 year period if the leaching rate the exists at 19 days is maintained over the 40 year period. It is evident from the test data and the analysis that a further reduction in the leaching rate is expected as the immersion time increases beyond 19 days. Since the Composite material has similar proportions of B_4C and binder the leaching rate obtained from the Plate material test data is concluded to be directly applicable to the Composite material.

Laboratory analysis of the Crystal River Composite material indicates that the minimum B^{10} concentration is 0.0125 gms of B^{10}/cm^2 and that the average concentration is 0.0150 gms of B^{10}/cm^2. These laboratory-determined concentrations do not include the B^{10} contained in either the B_2O_3 impurity in the B_4C powder or the boron constituent of the re-inforcing material. If it is assumed that all of the leachable boron comes from the B_4C material, the reduction in the minimum B^{10} concentration will be negligible (from 0.0125 to 0.01247 gms B^{10}/cm^2). The criticality margin for the Crystal River fuel racks was determined for a B^{10} concentration of 0.012 gms B^{10}/cm^2. It can, therefore, be concluded that the leaching of boron from the Composite material will not adversely affect the calculated margin even if significantly higher leaching rates are assumed.

Question 8

Describe the surveillance program that will be performed to show the continued presence of the boron in all of the boron plates over the complete life of the storage racks, and also show how a detected decrease of boron in the plates could be safely taken care of.

Response 8

The surveillance program for Crystal River is designed to permit samples of the B_4C Composite material to be periodically removed from the spent fuel pool and examined for changes in the physical and mechanical properties of the material. Stability of the physical properties (i.e. no significant changes in sample dimensions and weight) is indicative of a negligible change in the boron content of the Composite material. Sufficient samples will be available so that the mechanical properties (ultimate tensile strenght, modulus of elasticity) can be determined as a function of exposure on a regulatory scheduled basis throughout the life of the fuel racks.

The surveillance program will be designed to provide samples which are exposed simultaneously to the pool water and the gamma radiations and samples which are exposed to the pool water only. This is desirable since many storage locations will not see any gamma radiation for many years but will be continuously exposed to the pool environment.

While the major concern for the Crystal River Composite material is its long-term behavior in spent fuel storage pool water when simultaneously exposed to gamma radiation, it is obviously necessary for the material to also exhibit a satisfactory long-term behavior when it is exposed to pool water only. The optimum surveillance program would provide material behavior data for a given exposure well in advance of the time at which the material in the fuel racks reaches the same exposure. While it is possible for the surveillance samples to accumulate gamma exposure at a rate

significantly higher than the material in the fuel racks, it is clear that exposure to the pool water cannot be accelerated in any way by the Crystal River surveillance program. However, boron containing materials using the same binder are currently installed in other spent fuel pools (e.g. the Haddam Neck plant). The Haddam Neck surveillance program includes long-term exposure of the boron containing material to the pool water only and over nine months of exposure has been accumulated to date without any evidence that the material has been adversely affected. By the time the Crystal River fuel racks are installed, the Haddam Neck material will have accumulated almost two years of pool water exposure. Florida Power Corporation will monitor the results of the Haddam Neck surveillance program to obtain as much advance information on the behavior of the similar material as possible. It should be pointed out that Florida Power Corporation does not believe that long-term exposure of the Crystal River Composite material to the pool water will have any significant adverse effect on the material properties because material of similar composition to the organic binder in the Composite material have been in use for nearly fifty (50) years. The applications involve exposure of these materials to both salt and fresh water with no evidence of degradation due to the water environment.

Accelerated exposure to gamma radiation will be achieved by providing a sample holder which can be moved each outage to a new storage location surrounded by freshly discharged fuel. The sample holder will be similar in size and basic shape to a fuel assembly with the samples located around the periphery of the holder. The sample holder will be moved from storage location to location using available fuel handling equipment. The Composite material samples will be placed between stainless steel sheet and exposed to the pool water to simulate the geometry that exists in the fuel racks. Table 2 shows that the sample exposure level after five years will be at least a factor of 2 greater than the levels associated with fuel storage locations. The table also shows that an exposure level of 10^{11} Rad will be reached in 20 years for the surveillance samples while that level will not be reached during the 40 year life of the fuel rack.

The purpose of the surveillance program is to detect any unanticipated changes in the properties of the Composite material before such changes progress to the point where they adversely affect the boron content of the material and, hence, the criticality margin. Two mechanisms can be postulated for the loss of boron from the Composite material: 1) boron being leached from the material or 2) boron being removed in the form of grains of B_4C as the result of the loss of binder and/or reinforcing material integrity. The results of the leaching tests (see Response A-7) indicate that the loss of boron due to leaching will be negligible over the life of the racks. The results of the mechanical properties tests as well as the successful long-term commercial application of binder/reinforcing materials in aqueous environments indicate that the loss binder and/or reinforcing integrity is extremely unlikely (see Response A-6).

TABLE 2

GAMMA EXPOSURE LEVELS

	Fresh Fuel Stored for 40 years	Fresh Fuel Replacement Every 5 Years	Estimated Sample Exposure
0	0	0	0
1	6.7×10^{9}	6.7×10^{9}	5×10^{9}
2	9×10^{9}	9×10^{9}	1×10^{10}
3	1×10^{10}	1×10^{10}	1.5×10^{10}
5	1.2×10^{10}	1.2×10^{10}	2.5×10^{10}
10	1.4×10^{10}	2.4×10^{10}	5×10^{10}
15	1.5×10^{10}	3.6×10^{10}	7.5×10^{10}
20	1.6×10^{10}	4.8×10^{10}	1×10^{11}
25	1.7×10^{10}	6×10^{10}	1.25×10^{11}
30	1.8×10^{10}	7.2×10^{10}	1.5×10^{11}
35	1.8×10^{10}	8.4×10^{10}	1.75×10^{11}
40	1.8×10^{10}	9.6×10^{10}	2×10^{11}

An additional safeguard has been taken in the design of the Crystal River racks to minimize the effects of the Composite material becoming granular: the Composite material compartments in the storage cells are sifficiently tight to preclude the significant loss of B_4C grains even though the compartments are not watertight; the void regions in the poison compartments have been minimized to prevent a significant reduction in the level of the B_4C grams even if the loose B_4C grains fill the void regions.

In the unexpected event that the Crystal River surveillance program detected a loss of boron from the Composite material, Florida Power Corporation would develop a course of action which would prevent any condition resulting in an unacceptable criticality. The specific course of action would depend on the mechanism by which the boron is being lost, the rate at which it is being lost, the quantity of spent fuel stored in the spent fuel pools and the alternative storage options that would exist at the time of loss of boron was projected to become substantial. It should be emphasized that a significant quantity of boron can be lost from the Composite material before the criticality criterion (0.95 or less) is reached for spent fuel stored in the racks since the racks were designed to provide an acceptable criticality margin with fresh fuel, with a minimum specified B^{10} content and with no boron concentration in the spent fuel pool water.

Question 9

Provide data that shows that the high dose rates used for accelerated irradiation tests have the same effect on the boron plates as the lower dose rates that will be received in the spent fuel pool.

Response 9

Information available in the literature cautioned against the use of excessively high irradiation rates since data indicated that the associated irradiation times were too short to permit secondary processes to occur in organics which could influence the degradative process (see Reference 2).

Consequently the irradiation dose rate as well as the type of radiation were major considerations in developing and implementing the Carborundum test program to qualify the Crystal River B_4C Composite material. For the program, a gamma source (spent fuel from a test reactor) was selected which had gamma dose rates sufficiently high to accelerate the test program but not so much greater than the prototypical radiation levels associated with freshly discharged spent fuel as to invalidate the test results ($\sim 5 \times 10^7$ Rad/hr or less vs $\sim 10^6$ Rad/hr). The test program was

structured to include a gas generation test for the B_4C Plate material which has the same binder as the Composite material. The results of the gas generation test show that the rate of noncondensible gas generation in the B_4C plate material used in the gas generation test was essentially the same as that observed at Haddam Neck where B_4C Plates were irradiated by spent fuel over ~ 9 months to a level of $\sim 3 \times 10^9$ Rad. In addition the gas species and the associated volume magnitudes in the gas generation test were very similar to those determined for Haddam Neck. Since it is believed that gas generation is a sensitive indicator of the operational behavior of the binder, it has been concluded by Florida Power Corporation that the gamma dose rates utilized in the Carborundum qualification test program have the same effect on the Composite material as the lower dose rates that will be received in the Crystal River spent fuel pools.

References

1. Carborundum Report No. CBO-78-299, dated October 1978. This proprietary report has been submitted to NRC previously on November 3, 1978 by Northeast Utilities for Dockets 50-213 and 50-245.

2. Effects of Radiation on Materials and Components, Kircher and Bowman, Reinhold Publishing Corporation.

B. The following information is needed to supplement NES report number 81A0521, entitled "Nuclear Design Analysis Report for the Crystal River Unit 3 Spent Fuel Storage Racks", which was submitted on March 22, 1978:

Question 1

Provide the number of grams or uranium-235 per axial centimeter of fuel assembly that was used in your criticality calculations. We intend to incorporate this information as a Technical Specification limit on fuel assemblies that are to be placed in these high density storage racks. In this regard, it appears that your criticality calculations are based on the average mass of uranium dioxide per assembly, which is listed on Table 7.1 as 528 kilograms, rather than the maximum, which is listed as 536.94 kilograms. Is the uranium-235 loading that you used in the calculations high enough to permit the future storage of fuel assemblies with maximum loading and maximum enrichment without exceeding this intended technical specification limit?

Response 1

The criticality calculations reported in NES Report 81A0521 were based on a UO_2 loading of 528.0 kg per fuel assembly. This value represents the average fuel loading in 3.3 w/o Babcock and Wilcox fuel and corresponds to an axial density of U^{235} of 41.99 gms U^{235}/axial centimeter. However, the criticality value under worst case conditions (0.9356) is sufficiently below the NRC criticality criterion of 0.95 that the fuel storage racks can readily accommodate fuel having the maximum UO_2 loading of 536.94 kg. The sensitivity of K_{eff} to enrichment changes was investigated in NES Report 81A0521 (see section 7.3.3). The maximum UO_2 loading represents a 1.7% increase in the U^{235} concentration which corresponds to a change in k_{eff} of $+3.4 \times 10^{-3}$ for an infinite array of storage locations. Adding this increase to the above worst case criticality value (0.9356) results in a final k_{eff} value of 0.9390 for the maximum possible fuel loading. This value is still clearly below the NRC criticality criterion of 0.95. Therefore, it is possible to base the Crystal River Unit 3 Technical Specification limit on a maximum UO_2 loading of 536.94 kg.

Question 2

On Page 7-1 it is stated that the diffusion theory method was checked on a KENO calculation with 16-group Hansen-Roach cross sections. Provide documentation on the validity of this KENO calculational method for this type of boron plate calculation.

Response 2

The validity of using KENO to calculate criticality values for enriched Uranium fuel arrays which contain discrete boron plates have been studied extensively. The results, reported in NUREG-CR-0073, verify that the KENO calculational method is suitable for high density poison plate fuel racks of the Crystal River type.

Question 3

On page 7-2 it is stated that the nominal cell pitch of 10.5 inches will be maintained within $\pm 1/16$ inch. However, the dimensional specifications given in Table 7.4, which is entitled "Reference Case", show that the cell pitch for this calculation is

$$\frac{13.492\text{cm.} \times 2}{2.54 \text{ cm/inc.}} = 10.62 \text{ inches}$$

What is the caue of this apparent discrepancy?

Response 3

The cell pitch of 10.5 inches reported on page 7-2 of NES Report 81A0521 is consistent with the mesh locations shown in Table 7-4. The apparent discrepancy results from the EXTERMINATOR input format which requires that the outer boundaries of an array be located on the centerline between the outer two mesh points. For the case in question, the array boundaries lie between mesh points 1 and 2 and mesh points 18 and 19 for both the x and y dimensions. The portion of Table 7-4 titled "Dimension Specifications" presents the distance of subsequent mesh points from the first array boundary (located midway between mesh points 1 and 2). To check the size of the array from Table 7-4, it is necessary to subtract one-half of the distance between mesh points 18 and 19

$$\left(\frac{0.317 \text{ cm}}{2}\right)$$

from the distance for mesh point 19 (13.492 cm from the first boundary). The resulting dimension is 13.3335 cm which equals a half pitch value of 5.2494 inches. This, in turn, corresponds to a pitch of 10.4988 inches compared with the 10.5 inches specified on page 7-2.

Question 4

On Figure 7.2 on page 7-15 it appears that the boron region is too thick by a factor of ten. Is this correct?

Response 4

Figure 7-2 contains a typographical error. The correct value for the half-thickness of the boron region is 0.0375 inches.

Question 5

If it will be possible to place a fuel assembly between the outer periphery of the storage rack modules and the fuel pool walls, what will the closest possible distance to a stored fuel assembly be? What will the maximum neutron multiplication factor in the pool be when it is assumed that the outside assembly is in its most reactive position and when it is also assumed that the storage racks are filled with the most reactive fuel assemblies and there is no boron in the water?

Response 5

Structure is provided on the peripheral Crystal River fuel storage racks to prevent a fuel assembly from being placed closer than 6 inches to the storage locations adjacent to the north, south, and east pool walls. Calculations were performed which showed that the placement of a fuel assembly at 6 inches from fuel racks filled with the most reactive fuel will result in an increase in k_{eff} of +0.0032 (see Sections 5.2.1 and 7.4.2 of NES Report 81A0521). This change was included in developing the worst case criticality value presented in NES Report 81A0521 (0.9390).

C. The following information is needed to supplement Appendix C entitled "Cooling Considerations" of the January 9, 1978 submittal:

Question 1

Provide the design inlet temperature and the flow rate of the nuclear services closed cycle cooling water through the spent fuel pool heater exchanger.

Response 1

Design inlet temperature (shell) - 95°F

Design flow rate (shell) - 750,000 lb/hour

Question 2

Describe the procedure that would be used for aligning the Decay Heat Removal System for spent fuel pool cooling

Response 2

All crossover connection operations shall be performed within the plant limits and precaution considerations and governing operations and procedures. Initially the spent fuel cooling system would be isolated by closing SFV-9, 10, 43, 49, 51, and 54 and subsequently aligning the Decay Heat Removal System by opening SFV-87, 89, and DHV-7, 8, 39, 40, and 48.

Question 3

Provide the equilibrium spent fuel pool inlet and outlet water temperatures for a heat load of 8.75×10^6 BTU/hr and a spent fuel pool cooling system flow rate of 1500 gpm.

Response 3

Bulk water temperature inlet = 129°F

Bulk water temperature outlet = 117°F

The 129°F temperature results from the NRC postulated conditions of 8.75×10^6 BTU/hr and a cooling system flow rate of 1500 gpm. By comparison, the 125°F bulk water temperature, discussed in the response to Question 5 of Enclosure 1, results from an assumed heat load of 15.3×10^6 BTU/hr (10 succesive refuelings) and a cooling system flow rate of 300 gpm (two SFP coolers in operation) as reported in Table C-1 of GAI report number 1949.

ENCLOSURE 3

Question 1

Page 6 of the spent fuel pool structural analysis states that adjacent structural elements are analyzed to account for the restraint they contribute to the spent fuel pool structure. Provide justification that these elements retain their structural integrity under the imposed loads.

Response 1

A representative sample focusing upon the most critical structural elements surrounding the spent fuel pool including walls and floor slabs were examined to verify their structural integrity. The appropriate elements from the original finite element analysis study were checked to see that stresses due to mechanical and temperature loadings, combined according to specified loading combinations all were within limits established to preclude any loss of function. Stiffness was the major contribution by these adjacent members to the fuel pool analysis. Only the loading combination which was judged to be the most severe was examined.

For mechanical loads such as dead load, live load, hydrostatic, hydrodynamic and earthquake, the stresses in all the elements checked were very small.

For temperature loadings the stresses were high in most cases. For the case where the temperature induced stresses were compressive, the axial force and corresponding bending moment values at a location were all within the confines of the interaction diagram which is derived based on cross section geometry and material. For tensile axial forces, because the uncracked section assumption used in the analysis lacks validity due to concrete's inability to resist significant tension, the strain in the concrete using the given tensile force was computed to see if it exceeded that strain value needed to yield the main reinforcing steel. Using this procedure, it was found that there were no members which exhibited yielding reinforcing steel.

In conclusion, there were no members adjacent to the spent fuel pool which were stressed to such a level to cause them to exhibit a stiffness reduction or show any loss of their structural integrity.

Question 2

Discuss the effect of the water sloshing effects discussed in the rack analysis report on the fuel pool walls.

Response 2

As outlined in section 4.3.4(a) of the fuel pool structural analysis report, the water sloshing effects upon the fuel pool walls induced by seismic motion have been considered. Horizontal hydrodynamic forces have been applied to the walls and vertical hydrodynamic forces have been applied to the fuel pool floor in the analysis.

The lower portion of the water in the pool was considered contained and acted as a lumped mass in rigid contact with the pool wall and floor.

This mass of water experienced the same accelerations as the pool floor and walls. The top portion of the pool is free to slosh, thereby applying additional dynamic pressure to the pool walls. These forces were accounted for in the analysis by applying equivalent static loads to the pool walls and floor.

The stresses induced in the pool walls and floors due to these hydrodynamic effects have been included in the seismic effects (either E or E') when tabulating stresses for various load combinations.

Depending upon the location and direction of the earthquake, the hydrodynamic forces at a given point can induce large stresses when compared to the stresses caused by an earthquake with the pools empty. For example, at one point on the north wall of Pool A, the moment about the local y-axis due to a north-south hydrodynamic loading on Pool A only, is 2.32 ft-Kips/ft. At the same point the moment due to the north-south earthquake, ignoring hydrodynamic loading, is .936 ft-Kips/ft.

For the loading conditions involving hydrodynamic forces, the hydrodynamic stresses have been combined with stresses due to other loadings at many points on the structure, and it was found the stresses were all within the capacity of the existing structural members.

Question 3

Provide a direct comparison of loads used in pool structural analysis to those obtained from the rack seismic analysis.

Response 3

The table below presents a comparison of the fuel element and storage rack loadings used in the pool structural analysis and those used in the rack seismic analysis. Since the fuel pool analysis was done prior to the rack seismic analysis, the values listed for the pool analysis are representative of maximum values obtained during preliminary discussions with rack vendors.

	Pool Analysis	Rack Analysis
Submerged Weight of Rack and Fuel	2450 lb/ft^2	3148 lb/ft^2*
Vertical OBE Rack Acceleration	0.167g	0.033g
Horizontal OBE Rack Acceleration	0.54g	0.620g 1.050g
Vertical SSE Rack Acceleration	0.333g	0.067g
Horizontal SSE Rack Acceleration	1.08g	1.24g 2.10g

*Unsubmerged Weight
North-South Direction
East-West Direction

Since the seismic load is small compared with available section capacity, the apparent large difference between the rack acceleration values for the two analyses is not significant.

The use of the larger seismic values will not cause any of the structural elements to perform inelastically, thus allowing them to perform their load carrying functions with regard to strength and stiffness.

Question 4

Provide values in terms of g's used for the structural seismic loadings mentioned in 4.3.4(b) of the fuel pool structural analysis, also, indicate the directions of all applied seismic loads.

Response 4

Seismic forces due to the response of the spent fuel storage pool and the adjacent structural walls and slabs were obtained by multiplying the mass of the individual model element by the zero period floor response acceleration for elevation 143'-0". The values of these accelerations are:

	Horizontal	Vertical
OBE	0.075g	0.05g
SSE	0.150g	0.10g

Two-directional earthquakes were combined by absolute addition. Whenever stresses due to seismic loading were required for the stress summary of a particular loading combination, the following combinations of earthquake loading were applied. The case leading to the largest stress in an element in combination with other loadings governed the design of that element. The combinations of earthquake loading are:

Vertical + North

Vertical + South (=-North)

Vertical + East

Vertical + West (=-East)

Question 5

Section 5.7.1 of the fuel pool analysis states that "forces generated by the elastic analysis indicates that the existing reinforcement is insufficient..." Provide justification for the adequacy of the fuel pool under this loading condition.

Response 5

The axial tension and in-plane shear is due to temperature difference between the support wall and the pool wall. Since the region of

temperature transition is smaller than the size of the finite elements used in the analysis to model the structure, and only constant temperature can be input to the elements, there are large step changes in temperature of adjacent finite elements. Thus, the calculated thermal forces are much higher than the actual values. A refined local analysis using a larger amount of smaller elements to model the region will yield smaller thermal stresses which are closer to the actual value.

Even if assuming the overestimated thermal forces were true forces, the adequacy of the structure can still be demonstrated by considering the strain compatibility. For example, the most crtical principal tension is:

$$F = 290.6 \text{ kips}$$

$$\text{strain} = F/AE_c = \frac{290.6}{(12")(60)(3.32 \times 10^3)} = .122 \times 10^{-3}$$

which is much less than the yield strain of the reinforcement at $Ey = 1.62 \times 10^{-3}$. The apparent discrepancy of inadequate reinforcement by the force consideration and more than sufficient reinforcement by the strain consideration is due to the fact that uncracked concrete section is used in the analysis and totally cracked section is used to check reinforcement in force consideration. Since the localized high thermal force is self-limiting, the strain consideration should be used to check the reinforcement.

Question 6

Provide details showing areas where load and moment averaging is required to justify the structural adequacy of the fuel pool, and provide justification of the structural adequacy of all areas where the limits in the Standard Review Plan have not been met.

Response 6

Average element forces are used to:

a) justify vertical tension forces in north and south walls at junction with center wall

b) junction of slab elements with wall elements

c) justify tension forces in walls below the spent fuel pool.

It is a general practice to average the forces in adjacent elements in the reinforced concrete design subjected to mechanical loads. Since these are thermal forces, the section capacity can be justified by strain consideration without averaging the forces in the adjacent elements. For example, the most critical tension force is:

$$F = 374.2 \text{ kips}$$

$$\text{the strain} = F/AE_c = 0.1565 \times 10^{-3}$$

which is much less than the reinforcement yield strain $Ey = 1.62 \times 10^{-3}$.

Question 7

Quantify the statement in section 5.7.3, "However, in several isolated areas, generally whenever a sharp change in the mean temperature occurs between adjacent elements, very large stresses exist," and provide analytical justification for the structural adequacy of these areas.

Response 7

The sections can be justified by strain consideration as was explained in the answers to questions 5 and 6. The most critical tensile force in this case is:

$$F = 414.25 \text{ kips}$$

$$\text{the strain} = F/AE_c = .173 \times 10^{-3}$$

which is much less than the reinforcement yield strain, $Ey = 1.62 \times 10^{-3}$.

Question 8

Provide an explanation of the statement in the pool structural analysis given on page 28, "The embedment plates to which the fuel racks are attached were not evaluated for increased seismic loading".

Response 8

Based upon the new design of the free standing fuel racks, the embedment plates are not required to resist any rack overturning moment or horizontal force. These forces are resisted by horizontal sliding friction between fuel racks and the pool bottom liner plate. Hence, the only force to which the embedment plates are subjected is the vertical compressive force for which the plates have adequate capacity.

Question 9

Provide detailed drawings of the rack seismic bracing including the attachment to the pool wall.

Response 9

See Attachment 1 to Enclosure 3 for response to this question.

Question 10

The limit for load combination 5 given in section 5.2(b) of the rack structural analysis report should be 1.5S.

Response 10

Load combination 5, given in section 5.2(b) of the structural analysis report (NES Report 81A0522) includes the thermal and uplift loads and, therefore, is considered to be a factored load condition. The allowable limit for the factored load condition is taken as 1.6S, in accordance with section 3.8.4.II.5 of the Standard Review Plan. It should be noted that the stresses in the rack structure are significantly lower than either the 1.6S or the 1.5S limit.

Question 11

Discuss the effect of the gaps between adjacent rack assemblies and between the rack assembly and the seismic bracing on the seismic analysis results.

Response 11

See Attachment 1 of Enclosure 3 for response to this question.

Question 12

Provide the deflections of the rack base and fuel cans obtained from the seismic analysis.

Response 12

The maximum deflection of the rack base and fuel storage cells for various load combinations are given in Table 8.2(a) and 8.2(b) of the structural analysis report (NES Report 81A0522). Table 8.2(a) and 8.2(b) show that the maximum deflection in the rack base structure and fuel storage cells are approximately 0.0292 inches and 0.450 inches, respectively, for load combination 4 which includes the Safe Shutdown Earthquake.

Question 13

Provide justification that fuel pool liner will not develop leaks due to the applied loads including thermal loads. Also, provide justification that the stud bolts will not fail under the loadings.

Response 13

The pool liner is not considered as a structural member to take any mechanical load. Under thermal load, the liner is in compression; hence, will not develop leaks. The most critically stressed studs due to thermal load are those at the outside corners of the gate opening near the top of the middle wall, and at the top of the north and south walls. These are studs which are anchored into the concrete and welded to structural angles. The pool liner is welded to these angles. The calculated strain in the most severely loaded stud is 0.0041 which is greater than the yield strain of 0.00172. However, the ductility factor of 2.39 is well within the ductility range of the steel used for the studs and the studs will not fail under thermal load.

For studs attaching the liner to the concrete at locations away from the wall corners, the liner applies equal and opposite forces o the studs due to temperature loading. This type of loading causes negligible force on the studs.

Question 14

Explain how the spacer bars between each storage cell are considered in the seismic analysis model.

Response 14

The spacer bars have not been considered in the seismic analysis model since the spacer bars were designed to be relatively flexible and lightly attached to the storage cells. The spacer bars were not provided for structural reasons but to maintain storage cell alignment and spacing. The flexible, lightly attached design was used since it is only necessary for the major parts of the spacer bars to remain betwen storage cells during and after a seismic event in order to prevent any significant reduction in storage cell spacing.

Question 15

The mathematical model of the fuel rack assembly shows the feet are assumed anchored to the pool floor. How were the shear loads transmitted to the liner through the feet considered in the analysis of the linear integrity?

Response 15

The maximum effective frictional reaction force affecting the pool liner plate and racks support feet is 48,600 lbs.

This force is distributed over six feet for an average of 8,100 lbs for one foot. These values are taken from the "Structural Analysis Design Report", NES 81A0522.

The friction factor between steel and concrete is greater than that between steel and steel. Therefore, any force transmitted from the fuel rack support feet to the liner will go directly through the liner as shear into the concrete floor. No stress is present in the liner other than directly under the support, in addition to the small axial compressive stress equal to the frictional force divided by the support area. This produces an average shear stress of less than 1,000 psi which is well within allowable limits. Therefore, the pool floor liner is capable of withstanding the loads from the fuel racks.

Question 16

Provide results of an analysis considering a temperature gradient across the rack structure due to differential heating between a full and empty cell.

Response 16

Differential heating produced by full cells being adjacent to empty cells has negligible effect on the fuel storage racks. The rack base is totally unaffected by such differential heating since the base remains at the temperature of the pool water entering the rack. Differential heating, however, has the potential to produce different axial (vertical) expansions of the fuel storage cells. Based on the maximum temperature rise expectd in a full storage cell (see NES Report 81A0523), the full cell potentially could have a length no more than 0.025 inches greater than an adjacent empty storage cell. The only rack components which will see

this small differential expansion are the spacer bars which have been designed to be flexible and lightly attached to the storage cells. Consequently, the spacer bars will readily accommodate the small differential thermal expansions produced by full and empty cells. Even if it is assumed that the storage cell heating produces a thermal gradient solely across the two walls of the storage cell (inner cell wall at raised temperature and outer cell wall at pool temperature), the resulting axial stresses in the cell members are less than 4 ksi with the resulting axial load easily accommodated by the storage cell welds.

Question 17

Discuss the local effects of the impacting fuel assembly on the storage cell and the absorber material.

Response 17

For the Safe Shutdown Earthquake, the maximum fuel assembly impact load at each fuel assembly spacer grid location is approximately 193 pounds. The resulting maximum local bending stress in the storage cell wall at the cell wall/corner angle interface is approximately 10.0 ksi. This value is considerably smaller than the allowable bending stress of 27.0 ksi for stainless steel. Furthermore, since the fuel assembly spacer grid stiffness is much greater than the stiffness of the storage cell wall and the construction of the storage cell wall compartment is such that the impacting of the fuel assembly spaer grid will not cause any significant flexural deformation of the storage cell wall compartment, the flexible poison plates, which are located inside the cell wall compartments will not be subjectd to any significant local bending stresses. The poison plate will be subjected to a small local bearing stress of approximately 24.6 psi.

Question 18

Discuss how the effects of the impacting fuel assemblies were considered in the energy balance used to determine the maximum rack uplift during an SSE. Also, provide stability analysis for other rack assemblies.

Response 18

In the stability analysis of the storage racks presented in Appendix E of NES Report 81A0523, the effect of fuel assembly impact was considered by increasing the kinetic energy associated with the fuel assembly masses by a factor of 2, a conservative impact factor.

The results of the stability analysis are applicable to all of the rack types, since both the external kinetic energy and the potential energy required to lift the storage rack are proportional to the number of storage cells in the racks.

It should be emphasized that the external kinetic energy applied to the storage rack for the stability analysis in Appendix E is considerably greater than the value established by he Non-Linear Time History analysis performed to demonstrate rack adequacy for a free-standing installation. Consequently, the results of the stability analysis are conservative.

Question 19

Discuss the effects of a fuel assembly drop on a spent fuel storage cell which is directly above one of the rack feet. Include the effects on the fuel assembly.

Response 19

The effects of a fuel assembly drop on a fuel storage cell located directly above one of the fuel storage rack feet have been evaluated in Appendix F of NES Report 81A0523 (see pages 3 through 5).

The analysis indicates that the support feet base plate will be deformed and that there will be local crumbling of concrete under the support feet. However, the overall structural integrity of the storage rack and the leaktight integrity of the pool liner will be maintained.

The radiological consequences of a fuel handling accident (rupture of all fuel rods in one assembly) has been analyzed and found acceptable as described in section 15.0 of the NRC's Safety Evaluation Report.

Question 20

Appendix G of the rack structural analysis report gives a calculation of an adjusted frequency of a rack due to support flexibility in the upendor area. Provide a more detailed seismic analysis of the rack structure and seismic bracing.

Response 20

See Attachment 1 to Enclosure 3 for response to this question.

Question 21

Provide information on the heat treatment requirements and hardness testing results for the 17-4 PH stainless steel used for the leveling pads. Also provide information on the methods used for removing the film coating resulting from the heat treatment.

Response 21

The heat treatment requirements for the 17-4 PH stainless steel used in the fuel rack leveling pads shall be consistent with the requirements of ASTM Specification A-564 for Type 630. Specifically, the following requirements are being placed in the detailed rack drawings and fabrication specification which control the fabrication specification which control the fabrication of the fuel storage racks.

a. The 17-4 PH material shall conform to the requirements of ASTM Specification A-564 Type 630.

b. The 17-4 PH material shall be purchased in the solution-annealed condition, machined to its final configuration and then subjected to an age-hardening heat treatment.

c. The age-hardening heat treatment shall conform to the requirements of Tables 4 and 2A in ASTM Specification A-564 to achieve a tensile strength of 155 ksi (minimum) and a yield strength of 145 ksi (minimum). Specifically, each component shall be held at a temperature of 1025°F (552°C) for a period of four (4) hours and then allowed to air cool.

d. The heat treated components shall be tested and certified for a minimum hardness of 35 Rockwell C or 331 Brinnell. The use of either hardness measurement method is permissible. A minimum of 15% of the number of heat treated components (but not less than 1 component) shall be tested for each separate batch that is processed.

e. The surface film resulting from heat treating shall be removed from each component by using an aqueous solution of 10% HNO_3 - 2% HF at a temperature between 110 and 140°F. The component shall be exposed to this solution for a period not exceeding three minutes. Loosened scale may be removed by either the use of high pressure water or steam. Following scale removal the component shall be thoroughly rinsed in demineralized water and then dried with either clean air or lint free cloths. A uniform surface shall be considered evidence of a well-cleaned part.

Question 22

Discuss the fuel pool water chemistry and include the allowable concentrations of flourides, chlorides, and boron.

Response 22

The water chemistry of the spent fuel pools is maintained consistent with the requirements of the CR #3 Technical Specifications for the Reactor Coolant System with regard to fluorides, chlorides, and boron concentrations. The fluoride and chloride concentration limit for the fuel pools is $\leq$ .15 ppm. The boron concentration for the spent fuel pool is $\geq$ 1925 ppm.

Question 23

Dicuss the measures that will be taken to prevent outgassing from the B_4C absorber material in light of the current problems with stuck fuel assemblies.

Response 23

The B_4C poison material selected for the fuel storage racks has the potential of generating gas during its exposure to the spent fuel pool environment (namely, gamma radiation). In order to prevent any adverse effect from this gas generation the compartments in which the poison material is placed are open to the fuel pool at the top of the compartment. Consequently, it is not possible for the gas generation to produce any internal compartment pressure capable of deforming the storage cell. Therefore, the potential of stuck fuel assemblies due to gas generation has been precluded.

Attachment 1

Florida Power Corporation has determined that significant gaps exist between the fuel pool liner and the pool walls at the proposed seismic bracing elevation. These gaps, which exceed the thermal expansion gaps, are not consistent with the assumptions used in the seismic analyses as presented in NES Report 81A0523. The presence of such gaps significantly increase the impact loads applied to the pool walls and the fuel racks during a seismic event and the potential for pool liner rupture. It has been determined that extensive modifications to the spent fuel pool would be required to eliminate the gaps. Consequently, it was decided to investigate the feasibility of installing the Crystal River high density racks as free standing racks which are permitted to slide during a seismic event. Detailed non-linear time history analyses have been performed to establish the magnitude of sliding that will occur during the Safe Shutdown Earthquake. These analyses, presented in NES Report 81A0524, show that the maximum sliding displacement is 0.26 inches for a conservatively low coefficient of friction (0.25). The Crystal River high density racks shall be positioned so that the minimum clearance between fuel racks will be 1 inch and between the peripheral racks and fuel pool structure will be about 0.875 inches. These clearances are sufficient to preclude any collision between racks (even if racks move toward each other) and between the racks and pool structure during a seismic event.

Florida Power Corporation is, therefore, proposing to install the high density fuel racks described in the licensing amendment as free standing racks. Consequently, several NRC questions relating to seismically braced racks are no longer applicable: Items 9, 11, and 20 of Enclosure 3.