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AUTH.NAME AUTHOR AFFILIATION
WILLIAMS,J.W. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
VARGA,S.A. Operating Reactors Branch 1

SUBJECT: Forwards response to NRC 840906 request for addl info re proposed amend to expand spent fuel storage facility.Decay heat load analysis revised based on listed assumptions & bases.

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FLORIDA POWER & LIGHT COMPANY

October 5, 1984
L-84-264

Office of Nuclear Reactor Regulation
Attention: Mr. Steven A. Varga, Chief
Operating Reactor Branch #1
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Varga:

RE: TURKEY POINT UNITS 3 & 4
DOCKET NOS. 50-250 & 50-251
PROPOSED AMENDMENT TO
SPENT FUEL STORAGE FACILITY EXPANSION
ADDITIONAL INFORMATION

By letter dated September 6, 1984, the NRC requested that FPL provide additional infomation required to complete the Auxiliary Systems Branch review. The specific questions and responses are included as an attachment to this letter.

If additional information is needed, please contact us.

Very truly yours,

for J.W. Williams, Jr.
Group Vice President
Nuclear Energy

JWW/TCG/kgn

Attachment

cc: J.P. O'Reilly, Region II
Harold F. Reis, Esquire

A001
3/3

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT UNITS 3 & 4
SPENT FUEL STORAGE FACILITY EXPANSION
REQUEST FOR ADDITIONAL INFORMATION
AUXILIARY SYSTEMS QUESTIONS 5-13

Question No. 5: We have performed a spent fuel decay heat load calculation in accordance with the Standard Review Plan Section 9.1.3 and Branch Technical Position ASB 9-2 which does not agree with your calculation for the normal heat load conditions. Provide the results of a revised decay heat load analysis using the equations in the above referenced documents. Provide the results of the decay heat load analysis for the abnormal heat load case (one full core offload with the balance of the pool filled with half core refuelings). Based on these two analyses, provide a revised response to Question No. 1 which transmitted to you on May 11, 1984.

Response: The decay heat load analysis has been revised based on the following assumptions and bases:

1. The spent fuel pit heat exchanger is assumed to be available and operating 100 percent of the time (except for Case 1 where it is assumed to be lost at the time of addition of the last one-half core). Spent fuel pool cooling system flow is 2200 gpm (Ref. Turkey Point Updated FSAR Appendix 14E).

2. The heat transfer through the heat exchanger is calculated using the NTU-effectiveness method.

3. Component cooling water (CCW) flow to the spent fuel pit heat exchanger is 2800 gpm at a temperature of 100°F (Ref. Turkey Point Updated FSAR Appendix 14E and Section 9.3).

4. The mass of water in the SFP is conservatively assumed to be constant at its minimum value; that is, with the maximum volume deducted to account for the stored spent fuel.

5. The constant pressure specific heat, Cp, of the water in the spent fuel pit and the CCW is assumed to be constant with temperature at 1.0 BTU/lbm°F.

6. It is assumed that the spent fuel remains in the reactor for 150 hours before being placed in the spent fuel pool. For the full core off-load, this assumption is conservative since the full core would not be off loaded before 10 days (or 240 hours). This would cause the peak pool temperature to be slightly less than what was calculated.

7. The 10% uncertainty factor is applied to the calculations.

8. Irradiation times assumed were:

   11,000 effective full power hours - 1st offload
   22,000 effective full power hours - 2nd offload
   33,000 effective full power hours - 3rd through 6th offload
   38,000 effective full power hours - last 14 offloads

   Note: An irradiation time of 38,000 effective full power hours corresponds to fuel with a burnup of 50,000 MWD/MTU at 4.5 w/o enrichment.

9. The model used in ths calculation consists of the spent fuel pool (SFP) with the spent fuel stored in it and the SFP cooling system. Convective heat transfer off the surface of the pool and conductive heat transfer through the pool walls have been conservatively neglected.

10. 2200 MW_T Power was used for 100% reactor power.

11. The U_{avg} = 310 BTU/hr. ft. 2°F and heat transfer Area - 2000 ft^2 for the SFP Heat Exchanger.

12. Refueling Schedule considered for full core off-load.

Load #	Capacity	Decay Time	Time in Pool
19	1/2 core	150 hrs.	0
19	1/2 core	150 hrs.	0
18	1/2 core	36 days	29.75 days
17	1/2 core	1-1/2 yrs. + 150 hrs.	1-1/2 yrs.
16	1/2 core	3 yrs. + 150 hrs.	3
15	1/2 core	4-1/2 yrs. + 150 hrs.	4-1/2
14	1/2 core	6 yrs. + 150 hrs.	6
13	1/2 core	7-1/2 yrs. + 150 hrs.	7-1/2
12	1/2 core	9 yrs. + 150 hrs.	9
11	1/2 core	10-1/2 yrs. + 150 hrs.	10-1/2
10	1/2 core	12 yrs. + 150 hrs.	12
9	1/2 core	13-1/2 yrs. + 150 hrs.	13-1/2
8	1/2 core	15 yrs. + 150 hrs.	15
7	1/2 core	16-1/2 yrs. + 150 hrs.	16-1/2
6	1/3 core	18 yrs. + 150 hrs.	18
5	1/3 core	19 yrs. + 150 hrs.	19
4	1/3 core	20 yrs. + 150 hrs.	20
3	1/3 core	21 yrs. + 150 hrs.	21
2	1/3 core	22 yrs. + 150 hrs.	22
1	1/3 core	23 yrs. + 150 hrs.	23

13. Refueling Schedules considered for normal 1/2 core refueling.

Load #	Capacity	Decay Time	Time in Pool
20	1/2 core	150 hrs.	0
19	1/2 core	1-1/2 yrs. + 150 hrs.	1-1/2 yr.
18	1/2 core	3 yrs. + 150 hrs.	3
17	1/2 core	4-1/2 yrs. + 150 hrs.	4-1/2
16	1/2 core	6 yrs. + 150 hrs.	6
15	1/2 core	7-1/2 yrs. + 150 hrs.	7-1/2
14	1/2 core	9 yrs. + 150 hrs.	9
13	1/2 core	10-1/2 yrs. + 150 hrs.	10-1/2
12	1/2 core	12 yrs. + 150 hrs.	12
11	1/2 core	13-1/2 yrs. + 150 hrs.	13-1/2
10	1/2 core	15 yrs. + 150 hrs.	15
9	1/2 core	16-1/2 yrs. + 150 hrs.	16-1/2
8	1/2 core	18 yrs. + 150 hrs.	18
7	1/2 core	19-1/2 yrs. + 150 hrs.	19-1/2
6	1/3 core	21 yrs. + 150 hrs.	21
5	1/3 core	22 yrs. + 150 hrs.	22
4	1/3 core	23 yrs. + 150 hrs.	23
3	1/3 core	24 yrs. + 150 hrs.	24
2	1/3 core	25 yrs. + 150 hrs.	25
1	1/3 core	26 yrs. + 150 hrs.	26

Based on these assumptions, the following results are provided:

Half-Core (Normal Refueling)

Heat Load	1.79×10^7	BTU/hr
Maximum Pool Water Temperature	143	^{o}F
Time to Boiling After Loss of Cooling	7.6	hours
Evaporation Rate/Required Makeup	37	gpm

Full-Core Offload

Heat Load	3.50×10^7	BTU/hr
Maximum Pool Water Temperature	183	°F
Time to Boiling After Loss of Cooling	1.6	hours
Evaporation Rate/Required Makeup	72	gpm

Note: FPL normally replaces approximately 1/3 of the core at each refueling.

<u>Question No. 6:</u> The updated FSAR indicates that there is only one 7.96 MBTU/hr spent fuel pool heat exchanger. This is clearly undersized as your analysis indicates a 8.82 MBTU/hr heat load for the existing racks. Provide a commitment to install a second full capacity heat exchanger by the next refueling outage.

<u>Response:</u> The heat removal capability of the spent fuel pool (SFP) heat exchanger listed in FSAR Table 9.3-3 is 7.96×10^5 BTU/hr. based on a SFP water temperature of 120^oF (See note 2 for Table 9.3-3).

Since the temperature of the pool will rise above 120^oF due to the increased heat load, the heat removal capability of the heat exchanger will also change. The method utilized to evaluate the heat exchanger is based on the effectiveness of the heat exchanger in transferring a given amount of heat. Based on Reference 1, the heat exchanger effectiveness is defined as:

$$E = \frac{\text{actual heat transfer}}{\text{maximum possible heat transfer}}$$

The actual heat transfer may be computed by calculating either the energy lost by the hot fluid or the energy gained by the cold fluid, given by:

$$q = C_h (t_{h,in} - t_{h,out}) - C_c (t_{c,out} - t_{c,in})$$

where t_{in} = inlet temperature (^oF)

t_{out} = outlet temperature (^oF)

$C_h = (mC)_h$ hot fluid capacity rate $\left(\frac{BTU}{hr\ ^oF}\right)$

$C_c = (mC)_c$ cold fluid capacity rate $\left(\frac{BTU}{hr\ ^oF}\right)$

The maximum possible heat transfer for the heat exchanger could be attained if one of the fluids were to undergo a temperature change equal to the maximum temperature difference present in the heat exchanger. This is the difference between the inlet temperatures for the hot and cold fluids. The fluid which might undergo this maximum temperature difference is the one having the minimum capacity rate, since the energy balance requires that the energy received by one fluid be equal to that given up by the other fluid.

Thus, the maximum possible heat transfer is expressed as:

$$q_{max} = C_{min} (t_{h,in} - t_{c,in})$$

where C_{min} = the minimum of C_h and C_c

For the specific case of a shell and U tube heat exchanger, it may be shown that the effectiveness may be expressed as:

$$E = \frac{2.0}{(1 + (C_{min}/C_{max})) + (1 + (C_{min}/C_{max})^2)^{1/2}(1 + e^{-\Gamma})\,(1 - e^{-\Gamma})}$$

where C_{max} = the maximum of C_h and C_c

$$\Gamma = N_{tu}(1 + (C_{min}/C_{max})^2)^{1/2}$$

$$N_{tu} = UA/C_{min}$$

Then, from the characteristics of the heat exchanger and the hot and cold fluid flow rates, the effectiveness, E, may be calculated. Using the definition of effectiveness and knowing the hot and cold fluid inlet temperatures, the actual heat transfer is calculated.:

$$q = E\, q_{max}$$

Based on this method the heat removal capability for the SFP Heat Exchanger was modeled into the SFP cooling system analysis.

The maximum temperatures with the cooling system operating calculated for the 1/2 core and full core off-load cases are provided in the response to NRC Question 5. These temperatures represent the equilibrium point at which the heat removal capability of the heat exchanger matches the decay heat load being generated by the spent fuel. These temperatures are considered acceptable and, therefore, no modifcations are required to the SFP cooling system.

Ref.1: Kays, Wm. and A. L. London; Compact Heat Exchangers; 2nd edition; McGraw-Hill Book Company.

Question No. 7: The updated FSAR is not clear. Either 1) verify that there is an interconnection between the spent fuel pool and the RHR system and provide P&ID(s) which show the interconnection, 2) commit to provide the interconnection in (1) by the next refueling or 3) provide the results of an analysis which shows that no offsite dose limits and personnel exposure limits will be exceeded by allowing the pool to boil with makeup from only the seismic Category I source(s).

Response: No interconnections exist between the spent fuel pool (SFP) cooling and RHR systems.

Given the loss of all non-seismic sources of makeup water to the spent fuel pool, the spent fuel pool can be filled from the discharge of a high head safety injection pump. The source of water is either/both refueling water storage tanks via fire hose to either/both fuel pools.

Upon loss of all non-seismic sources of makeup water to the spent fuel pool(s), fully isolatable spool piece(s) in the HHSI pump test line(s) will be removed, interface fitting(s) installed, and fire hose connected and routed to the spent fuel pool(s).

1. All fittings required for this evolution will be purchased or fabricated and dedicated to this application.

2. Applicable plant procedures will be revised to reflect this source of makeup water and the method to accomplish the flow path.

A conservative analysis has been performed to determine the radiological consequences of a postulated spent fuel pool boiling event with makeup from only the Seismic Class I RWST. This analysis is consistent with the methodology and assumptions utilized in a similar pool boiling calculation performed for the

Limerick plant (Reference 1). The Limerick pool boiling analysis was reviewed by NRC and found acceptable in Reference 2.

The following assumptions were used for the pool boiling analysis:

a. The saturation noble gas and iodine inventories in the core are based on a power level of 2300 MWt with an initial enrichment of 4.5 $^{w}/_{o}$ and a discharge burnup of 50,000 MWD/MTU.

b. The SFP cooling systems for both pools (Units 3 and 4) fail simultaneously, one containing a full-core offload of 157 assemblies decayed for 150 hours, and the other pool containing a half-core from the last refueling decayed for 150 hours.

c. Pool boiling occurs instantaneously upon loss of SFP cooling; no credit is taken for decay during the pool heatup period.

d. 1% of the fuel rods in the core are defective. The 1/2-core from the last refueling is assumed to contain the defective 1% of the fuel rods from that core.

e. The gap activity consists of 10% of the total noble gases except Kr-85, 30% of the Kr-85 activity, and 10% of the total radioactive iodine contained in the fuel rods.

f. Because of their short decay-times, I and Xe in fuel from past refuelings are neglible.

g. Activity in the SFP water at the initiation of boiling is neglible compared to the activity released from the fuel during pool boiling.

h. The iodine and noble gas leakage rates from the fuel rods are 1.3E-8 sec^{-1} and 6.5E-8 sec^{-1}, respectively (Reference 3). These are the full power design fuel leak rates.

i. Although there is no data to support the phenomena in the SFP boiling scenario, iodine spiking factors of up to 50 are analyzed. In general, spiking has been observed during abrupt temperature and pressure transients associated with startup and shutdown, but such significant spiking effects would not be expected during the gradual temperature change that would be associated with a loss of SFP cooling. Since the temperature of the fuel during boiling is expected to be well below the normal reactor core operating temperature, the use of the full-power leakage rate is considered to be conservative.

j. Activity released from the fuel is uniformly mixed in the SFP water volume.

k. The pool water level is at El. 38' with continuous makeup capability (see the discussion below). This height corresponds to the elevation where the SFP cooling system lower suction line penetrates the SFP wall.

l. The iodine partition factor at the pool surface is 0.1.

m. The activity release rate from both pools is conservatively based on an evaporation (boiloff) rate for a full-core offload (see the response to Question No. 5).

n. All activity escaping from the pool is instantaneously released at ground level to the atmosphere without filtration or condensation in the ventilation system.

o. The atmospheric dispersion (X/Q) factors for dilution are taken from Section 14.3.5 of the Updated FSAR.

As shown in Table 1, the offsite dose consequences of a postulated pool boiling event are a small fraction of 10CFR100 limits.

References:

1. Final Safety Analysis Report, Limerick Generating Station Units 1 and 2, Volume 10, Section 9.1.

2. Safety Evaluation Report Related to the Operation of Limerick Generating Station Units 1 and 2, NUREG-0991, August 1983.

3. Source Term Data for Westinghouse Pressurized Water Reactors, WCAP-8253, July 1975.

Table I

Results of Spent Fuel Pool Boiling Analysis

Site Boundary 2-Hour Dose (Rem)

	Spike = 1	Spike = 50
Thyroid	5.7E-3	2.8E-1
Whole-Body	1.3E-5	1.8E-4

LPZ 30-Day Dose (Rem)

	Spike = 1	Spike = 50
Thyroid	1.4E-2	5.6E-1
Whole-Body	2.8E-5	1.8E-4

Question No. 8: The Updated FSAR indicates that the spent fuel pool cooling system is designed for a maximum temperature of 200°F and the storage capacity submittal indicates that the spent fuel pool is designed for a temperature of 150°F. Provide a discussion of the effects of a sustained pool water temperature of 212°F on the pool and on the cooling system. Provide the anticipated time until failure of the pool structure and the effects of the anticipated failure of the pool structure and the effects of the anticipated failure.

Response: The Updated FSAR has been revised to state that a temperature gradient (not a temperature difference) of 150°F (180°F inside pool and 30°F outside air) was considered in the structural analysis of the pool. Additional structural analyses have been conducted on the pool for a temperature gradient reflecting a 212°F water temperature and 30°F air temperature. Except for the changes in water temperature, the analysis was identical to that described in response to NRC Questions 9, 10, and 11 (questions submitted to Florida Power and Light via NRC letter of August 13, 1984). The analysis conservatively assumes that sufficient time elapses to allow thermal equilibrium to be reached; the results are therefore independent of time.

For the load case which includes thermal considerations but ignores the seismic event, the pool would be expected to remain at 212°F for only a short time before corrective action would result in reduced temperatues. During this time period, the likelihood of a seismic event, particularly at this site, is considered extremely remote. As shown in Table A, this load case results in stresses which are within the original design criteria allowables. When both seismic and thermal conditions are assumed to occur simultaneously, there occur localized instances of reinforcing steel stresses slightly exceeding the allowable stress of 36 ksi (see Table A). However, these minor localized occurrences (which take place only if seismic and thermal accident conditions occur simultaneously) cannot be construed as causing a loss of function or loss of structural integrity. The criteria included in the Updated FSAR (Appendix 5A, Section II) recognizes this fact by allowing such types of occurrences (including localized yielding) under certain conditions. Additionally, Reference 1 recognizes the self-limiting nature of thermal stress in reinforced concrete and allows reinforcing strains in excess of yield strain under thermal accident conditions. Although this reference does not apply to Turkey Point Units 3&4 as a design basis, it does represent an approach to thermal loads considered acceptable by the engineering community.

Liner plate integrity was investigated in a separate analysis. This analysis conservatively considered both the difference between the concrete temperature (average of 212°F and 30°F) and temperature of the liner plate (212°F) as well as the differences in thermal coefficients of expansion of the two materials. The analysis evaluated the liner plate, as well as stresses in the welds and embeds associated with it. The results of the analysis showed that there will be no loss of function.

In conclusion, both structural integrity and pool function will be maintained for an indefinite period of time if the water temperature in the pool were to be maintained at 212°F. In addition, the fuel pool cooling system and components have been found suitable for operation at 212°F.

Reference:

1. ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsection CC-3000 (including Summer 1981 Addenda)

TABLE A

SPENT FUEL POOL
CONTROLLING STRESSES FOR
212°F WATER TEMPERATURE

LOCATION	LOAD COMBINATION 1.25 (D+P+L)+T_{212} f_s ksi	f'_s ksi	ϕFy / Rebar Stress (1)	1.25 (D+P+L) E' +T_{212} f_s ksi	f'_s ksi	ϕFy / Rebar Stress (1)
Base Mat	12.0	-	3.00	6.3(2)	-	5.71
East Wall (Canal)	17.4	-	2.07	fv = 142 psi(3)		1.04(3)
East Wall (Pool)	36.0	-8.1	1.00	39.5	-10.3	ε=1.1
North Wall	30.5	-1.5	1.18	32.1	-3.8	1.12
South Wall	34.9	1.9	1.03	38.9	1.0	ε=1.08
Middle Wall	9.5	9.4	3.79	9.5	7.7	3.79

f_s = Stress in tension steel
f'_s = Stress in compression steel
f_v = Concrete shear stress

NOTES: (1) Based on a cracked analysis, reinforcing steel stress is obtained directly.

Due to the self-relieving nature of thermal loads on reinforced concrete, the ratio of maximum moment capacity to actual moment cannot be uniquely determined. As an alternative, the ratio ϕ Fy to computed reinforcing steel stress is provided. Since structural integrity is maintained beyond the allowable stress for thermal loading, the actual safety factor is greater than the ratio reported.

In those cases where the computed reinforcing stress exceeds 36 ksi, a strain ratio is provided were:

$$\varepsilon = \frac{\varepsilon_s}{\phi\varepsilon_y}$$

ε_s = reinforcing steel strain

$\phi\varepsilon_y$ = reinforcing steel strain at allowable stress

(2) This stress represents the maximum stress found in the top layer of reinforcing steel in the thinner center section of the base mat. The top steel in this area is important for transfer of the tensile loads imposed by the lateral water pressure from the pool. The bottom steel in the center portion of the base mat of the pool is used primarily for crack control. Since the base mat rests directly on competent fill material, stresses in this bottom (secondary) steel resulting from thermal loads have no adverse effect on the ability of the pool to transfer load. Therefore, the stress in the bottom steel is not included in Table A.

(3) This section occurs in the 3 foot wide by 18 inch thick section of the east wall between the two canal walls. Because of the short span of this section, and the large ratio of section thickness to span length, the section does not resist loads in the fashion of a shallow beam; shear stresses control the section capacity. Since shear stirrups are provided, the allowable shear stress in the concrete exceeds 148 psi. The ratio provided in Table A is that of allowable shear stress divided by f_v. The reinforcing steel on the outside face of this section is used only for crack control and is not needed to resist mechanical loads. Therefore, the flexural stresses in this reinforcing steel are not included in Table A.

Question No. 9: The submittal is unclear as to the intended use of the new fuel storage facility. Is it your intention to convert the new fuel storage facility for storage of spent fuel? Provide a discussion of your intended use of the new fuel storage facility and any changes between the existing system and the proposed system.

Response: It is not our intention to convert the new fuel storage facility for storage of spent fuel. All modifications under the subject submittal are restricted to the present spent fuel storage facility.

<u>Question No. 10:</u> The submittal stated that the temporary crane, racks, and staging platform will have to be carried over the exclusion area identified in the drawings submitted in response to NUREG-0612. Therefore, provide the safe load path drawings requested in our Question No. 4.

<u>Response:</u> As noted in the Safety Analysis Report, the lifting of the racks, temporary construction crane, and staging platform will necessitate passage over portions of the exclusion areas identified in Reference 1. The areas involved are (1) the area between the cask washdown area and the spent fuel pool east wall and (2) the spent fuel pool itself. With the exception of these areas, heavy loads will not be handled over the exclusion areas identified in Reference 1. The re-racking operation cannot be completed without carrying the racks, temporary construction crane, and staging platform over the exclusion areas identified above. Therefore, it is not possible to provide safe load path drawings within these areas.

As noted in previous responses (see Reference 2), the effects of a postulated load (cask) drop in the area between the cask washdown area and the spent fuel pool east wall have been previously addressed and found acceptable in correspondence with the NRC regarding the transfer of spent fuel between Turkey Point Units 3 and 4. The postulated drop of any item within the spent fuel pool with the normal water level would be enveloped by the cask drop analysis addressed in the SAR. In this analysis, it was assumed that the number of fuel assemblies damaged would be equal to (a) the number offloaded during a normal refueling plus the remainder of the pool filled with discharged assemblies from previous refuelings, or (b) a full core offload plus the remainder of the pool refilled with discharged assemblies from previous refuelings. An analysis has also been performed for a construction accident with reduced pool water level, as discussed in the response to Question No. 13. In both cases, the consequences of an accident have been found to be acceptable.

References:

1. Letter for R. E. Uhrig, FPL to S. A. Varga, NRC, L-82-346, dated August 10, 1982

2. Letter from J. W. Williams, Jr. FPL, to S. A. Varga, NRC, L-84-165, dated July 2, 1984.

<u>Question No. 11:</u> Verify that the procedures will require that the transportation of loads follow the safe load paths identified on the drawings that you will provide in the response to Question Number 10.

<u>Response:</u> As discussed in the response to Question No. 10, it is not possible to provide safe load paths for the temporary crane, racks and staging platform. However, all rigging and operations for the installation of these items will be in accordance with the requirements of NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants."

<u>Question No. 12:</u> Will any special lifting devices be used? For each special lifting device, provide a comparison to Guideline 4 of NUREG-0612, "Special Lifting Devices," and verify that it is single failure proof.

<u>Response:</u> The lifting device for the racks is a special lifting device. This lifting device has been designed for a load of six (6) times the combined static and dynamic loads from the racks, plus intervening components of the special lifting device, without generating a combined shear stress or maximum tensile stress at any point in the device in excess of the corresponding material minimum yield strength. It is also capable of resisting ten (10) times the above load without exceeding the ultimate strength of the materials. The above design factors are in accordance with the requirements of Guideline 4 of NUREG 0612, for special lifting devices which are not single failure proof. All other applicable requirements of this guideline have been met. (The special lifting device satisfies the guidelines of ANSI N14.6 - 1978)

Procedures will require that Florida Power & Light personnel sign off on the proper installation of the special lifting device prior to each use.

Question No. 13: The new racks will hold more spent fuel than the existing racks, therefore it is not clear that a cask drop accident with the new racks will be bounded by a cask drop accident with the old racks. Provide a discussion of the cask drop accident with new racks.

Response: The radiological consequences of a cask drop accident with the new storage racks are bounded by a cask drop accident with the existing racks due to the increased amount of decay time specified prior to cask handling operations. For the existing racks, Technical Specification 3.12, Cask Handling, presently requires a minimum of 1000 hours of decay for all spent fuel stored in the pool prior to cask handling operations. For the new spent fuel racks with increased fuel storage capacity, this decay time will be increased to 1525 hours (see proposed Technical Specification 3.12 and SAR Section 5.3.1.2). By requiring spent fuel decay time for the new spent fuel racks to be a minimum of 1525 hours prior to moving a spent fuel cask into the spent fuel pit, the potential offsite doses from a cask drop accident in the SFP (see SAR Table 5-8) will not increase over those previously evaluated for the existing racks. The calculated doses for a cask drop are well within 10 CFR 100 limits.

In addition, to ensure that potential offsite doses from a construction accident during rack replacement are less than those from a cask drop accident, the reracking operation will take place no sooner than 2150 hours after shutdown for the last batch of spent fuel placed in the SFP. This increased decay time is required since the water level in the SFP will be reduced approximately 8 feet during rack handling operations which results in a reduced pool decontamination factor per Regulatory Guide 1.25.