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SUBJECT: Forwards revised responses to Questions 2, 6 & 7 re replacement of spent fuel racks, per NRC 870716 request for addl info & subsequent discussions w/util.

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FPL

DECEMBER 23 1987
L-87-537

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Re: St. Lucie Unit 1
Docket No. 50-335
Spent Fuel Rerack

By letter L-87-245, dated June 12, 1987, Florida Power & Light Company (FPL) submitted a proposed license amendment to permit replacement of the spent fuel pool racks at St. Lucie Unit 1 to ensure that sufficient future capacity exists for storage of spent fuel.

By letter dated July 16, 1987 (E. G. Tourigny to C. O. Woody) the NRC Staff requested additional information in the area of plant systems it needed to continue its review of this proposed license amendment. By letters L-87-374, dated September 8, 1987, and L-87-419, dated October 19, 1987, FPL submitted responses to this request. As a result of subsequent discussions with the NRC, it was determined that FPL's responses to questions 2, 6, and 7 would need to be revised. Attached are FPL's revised responses.

If additional information is required, please contact us.

Very truly yours,

C. O. Woody
Executive Vice President

COW/EJW/gp

Attachments

cc: Dr. J. Nelson Grace, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant

A001
1/1

QUESTION #2

QUESTION: The submittal specifies that the spent fuel pool cooling system consists, in part, of two spent fuel pool cooling system pumps and one heat exchanger. Since all heat exchangers require maintenance at some point in time, it is not clear how the spent fuel pool will be cooled during the time that the single spent fuel pool cooling system heat exchanger is out of service for maintenance. Provide a discussion of how the spent fuel pool is cooled and the temperature is maintained below the Technical Specification temperature limit for the following conditions:

1. The heat exchanger is out of service for plugging of failed tubes.

2. The heat exchanger is out of service for cleaning of the tubes.

3. The heat exchanger is out of service for replacement.

RESPONSE: Due to the low pressure, low temperature service environment, and the carefully controlled water chemistry, (of both the CCW and SFPCS) the SFP heat exchanger will only infrequently require maintenance. Maintenance operation such as cleaning and tube plugging can normally be scheduled in advance for periods of low pool decay heat loads (just before a normal refueling). The spent fuel pool heat exchanger is designed for the life of the plant, therefore its replacement is not postulated. However, in the extremely unlikely event that replacement is required, alternate means of providing cooling to the SFP could be implemented during replacement.

If there were no cooling provided to the SFP such that boiling were to occur, redundant sources of makeup water to the SFP are available, as discussed in St Lucie Unit 1 FSAR Amendment No 6, Section 9.1.3.4 and in Section 3.2.3 of the St Lucie Plant Unit No 1 Spent Fuel Storage Facility Modification, Safety Analysis Report, transmitted via letter L-87-245 dated June 12, 1987. Boiling of the water in the SFP is adddressed in Section 9.1.3.4 of the FSAR and in Section 3.2.2.4 of the Safety Analysis Report. (There is no Technical Specification temperature limit for the SFP.)

Attachment 6 contains the Spent Fuel Pool Boiling Analysis. The results of this analysis demonstrate that the maximum calculated thyroid dose is much less than 1% of the 10 CFR 100 limit of 300 rem and the whole body and skin does are completely insignificant. Therefore, no off-site dose limits will be exceeded if boiling occurs in the spent fuel pool.

QUESTION #6

QUESTION: For each step in the replacement of the racks, provide drawings which show the location of the spent fuel and the movement of the fuel which demonstrates that no heavy load will be carried over spent fuel or over any rack which contains spent fuel. (Note: This information can be placed on the set of drawings provided in response to question 5).

RESPONSE: The attached drawings show the various locations of the spent fuel during the rack installation. Described below is the sequence of rack removal and installation.

1. Relocate all the fuel assemblies presently stored in existing Racks 1 and 2. Following the removal plan (see Question #5), remove Racks 1 and 2 from the pool as shown in Figure 1.

2. Following the installation plan (see Question #5), install new Racks F1, G2 and G1 as shown in Figure 2.

3. Relocate all the fuel assemblies stored in existing Racks 3, 4, 5 and 6 to the southernmost cells of new racks F1, G2 and G1 as shown in Figure 3.

4. Following the removal plan (see Question #5), remove existing Racks 3, 4, 5 and 6 as shown in Figure 3.

5. Following the installation plan (see Question #5), install new Racks D2, C3 and C1 as shown in Figure 4.

6. Relocate all the fuel assemblies stored in existing Racks 7, 8, 9, 10, 13 and 14 into new Racks D2, C3 and C1 as shown in Figure 5.

7. Following the removal plan (see Question #5), remove existing Racks 7, 8, 9, 10, 13 and 14 as shown in Figure 5.

8. Following the installation plan (see Question #5), install new Racks D3, B2 and C4 as shown in Figure 6.

9. Relocate all the fuel assemblies stored in existing Racks 11 and 12 into new Rack D3 as shown in Figure 7.

10. Following the removal plan (see Question #5), remove existing Racks 11 and 12 as shown in Figure 7.

11. Following the installation plan (see Question #5), install new racks B1, C2, A2, A1, D1, H1, E2 and E1 as shown in Figure 8.

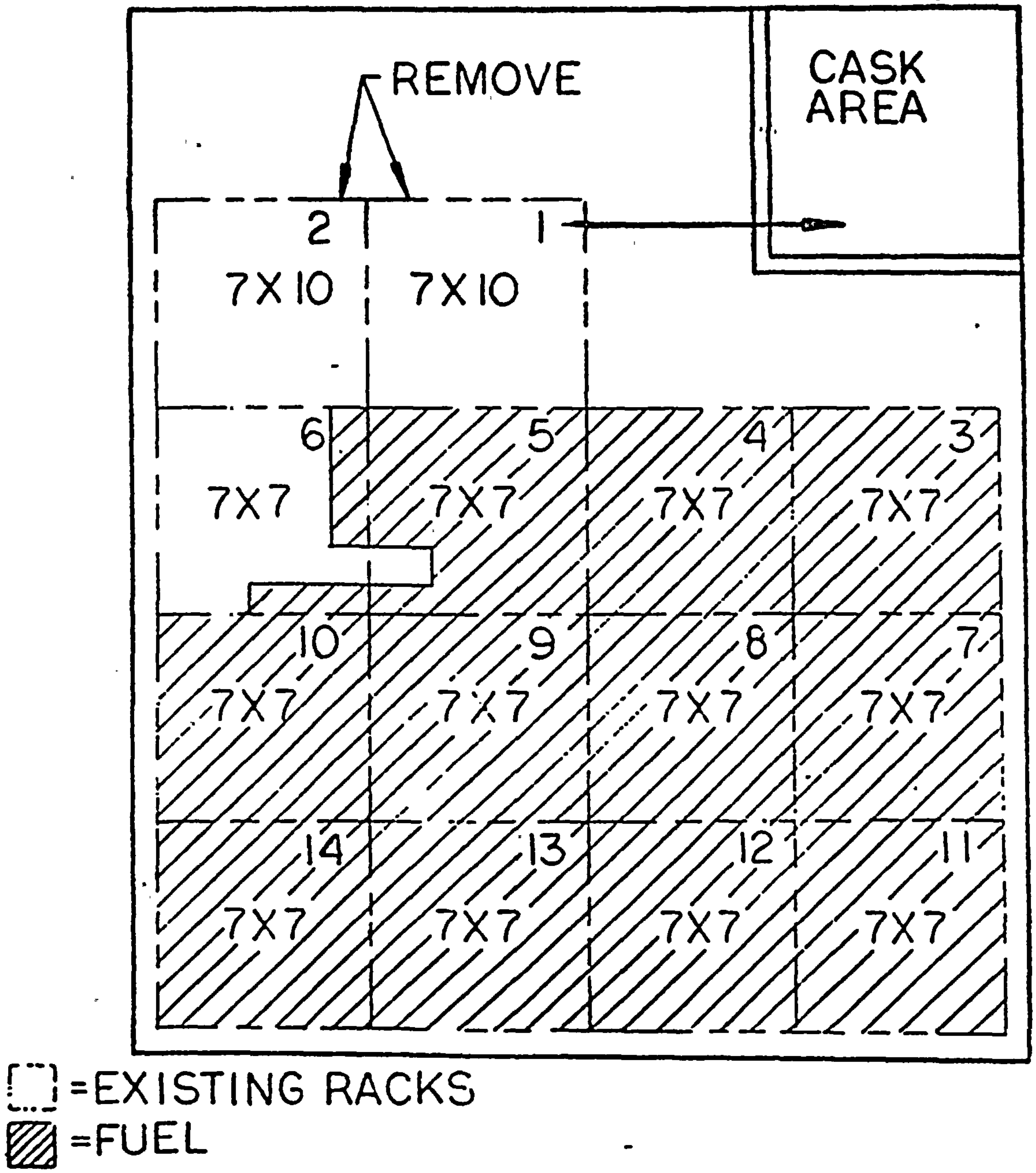

FIGURE 1 (REV 1)

nuclear utility construction

TITLE: SEQUENCE OF RACK REMOVAL AND INSTALLATION

ENCLOSURE: 1 OF 8

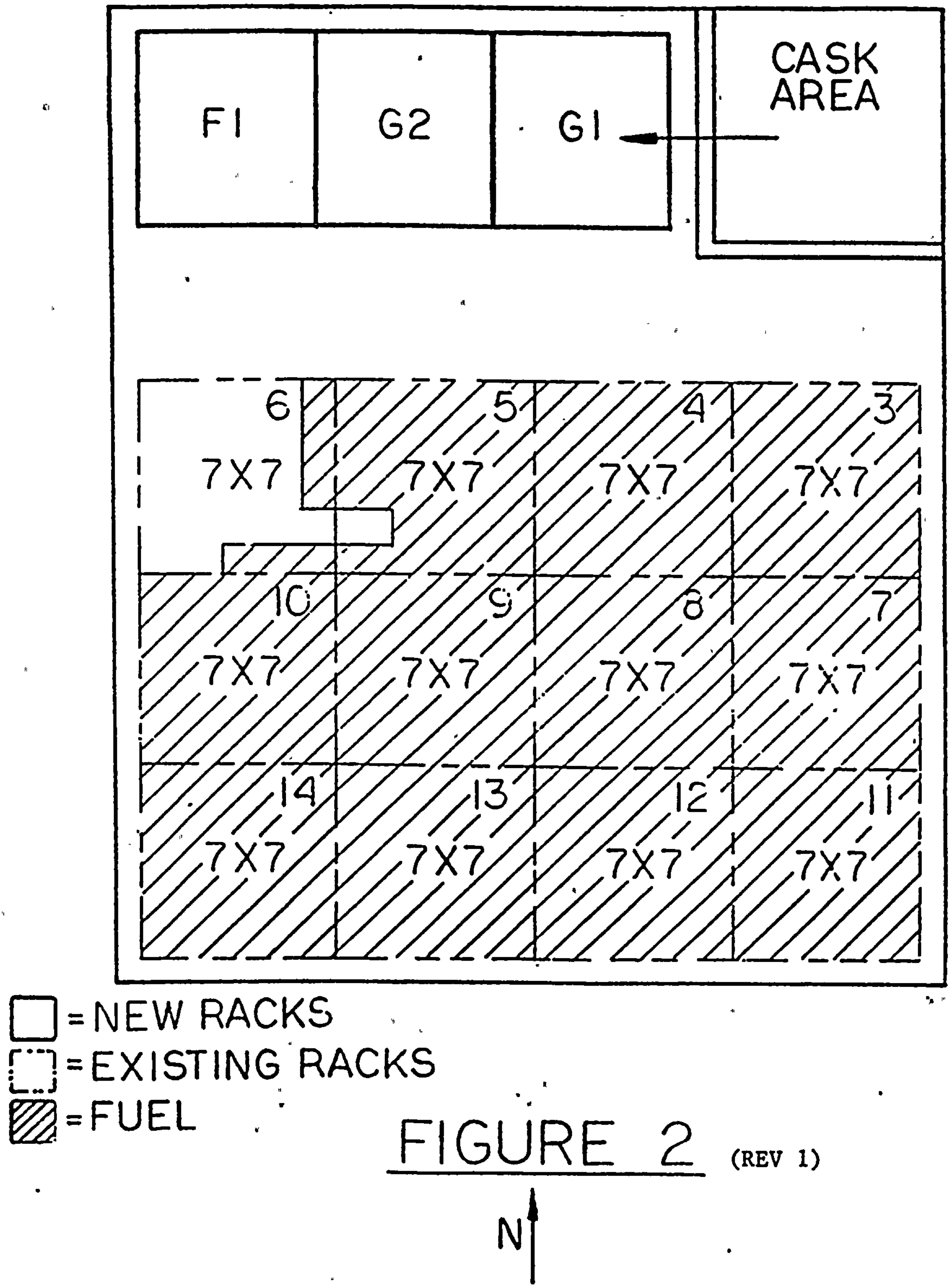

FIGURE 2 (REV 1)

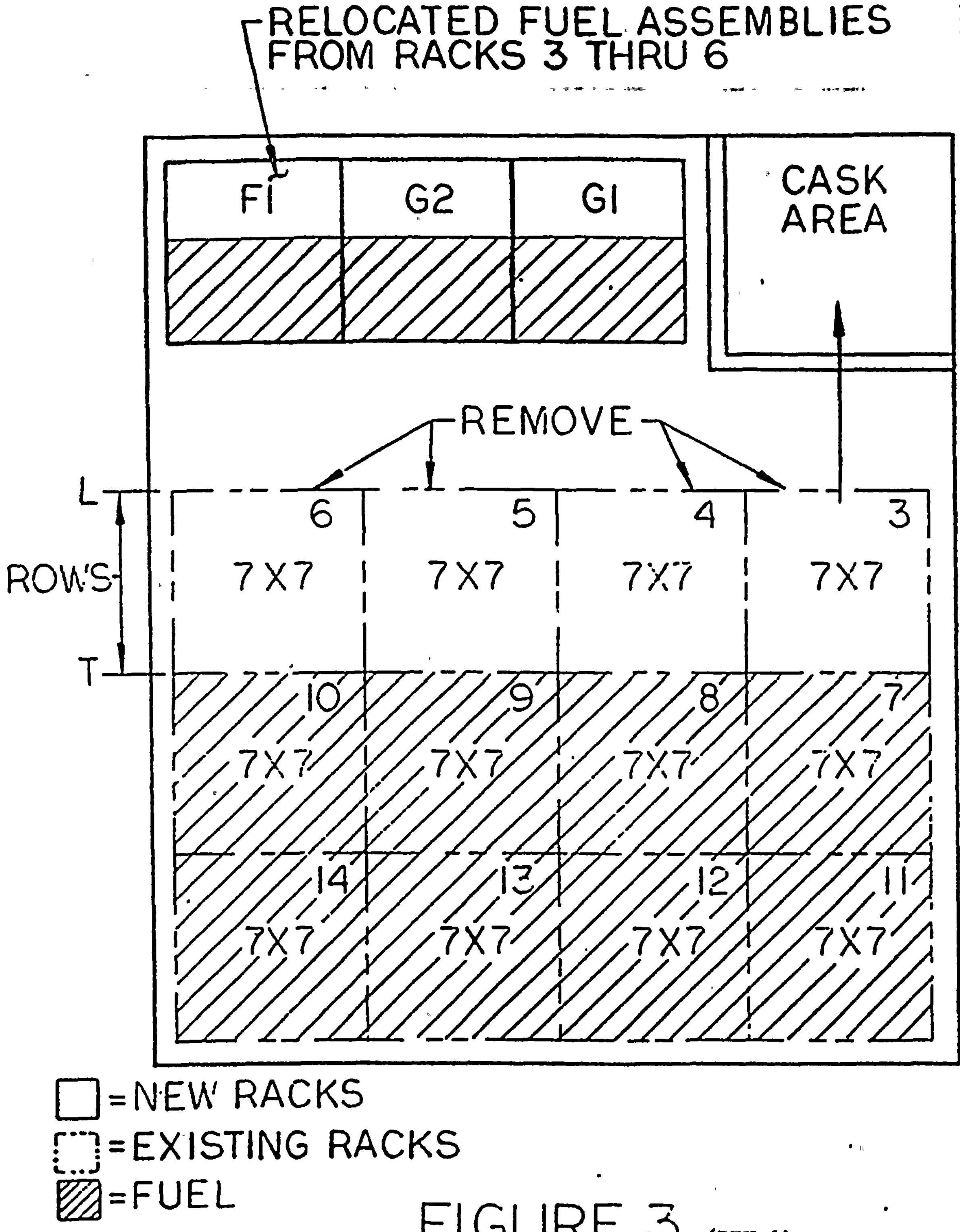

FIGURE 3 (REV 1)

nuclear utility construction

TITLE: SEQUENCE OF RACK REMOVAL AND INSTALLATION

ENCLOSURE: 3 OF 8

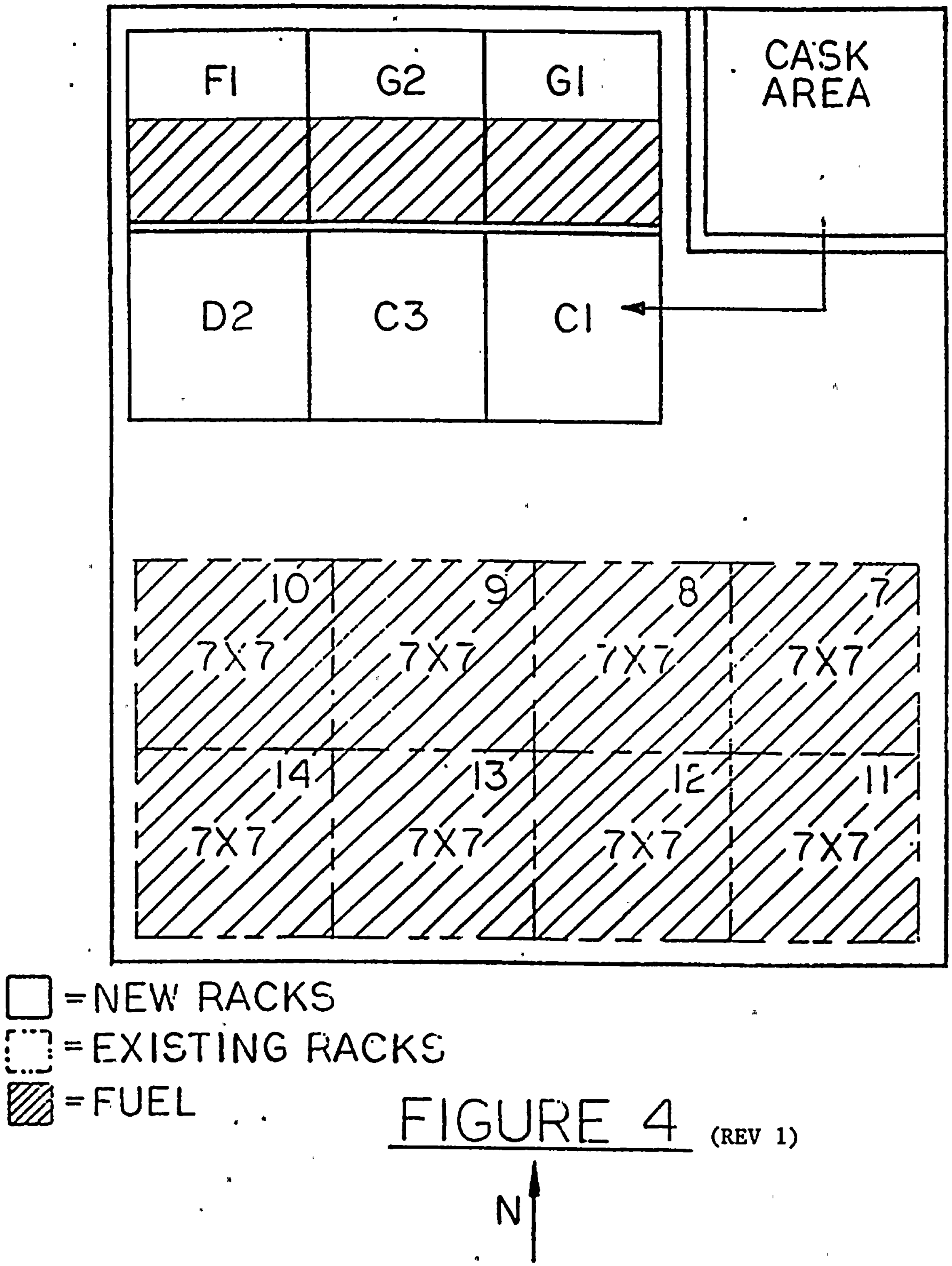

FIGURE 4 (REV 1)

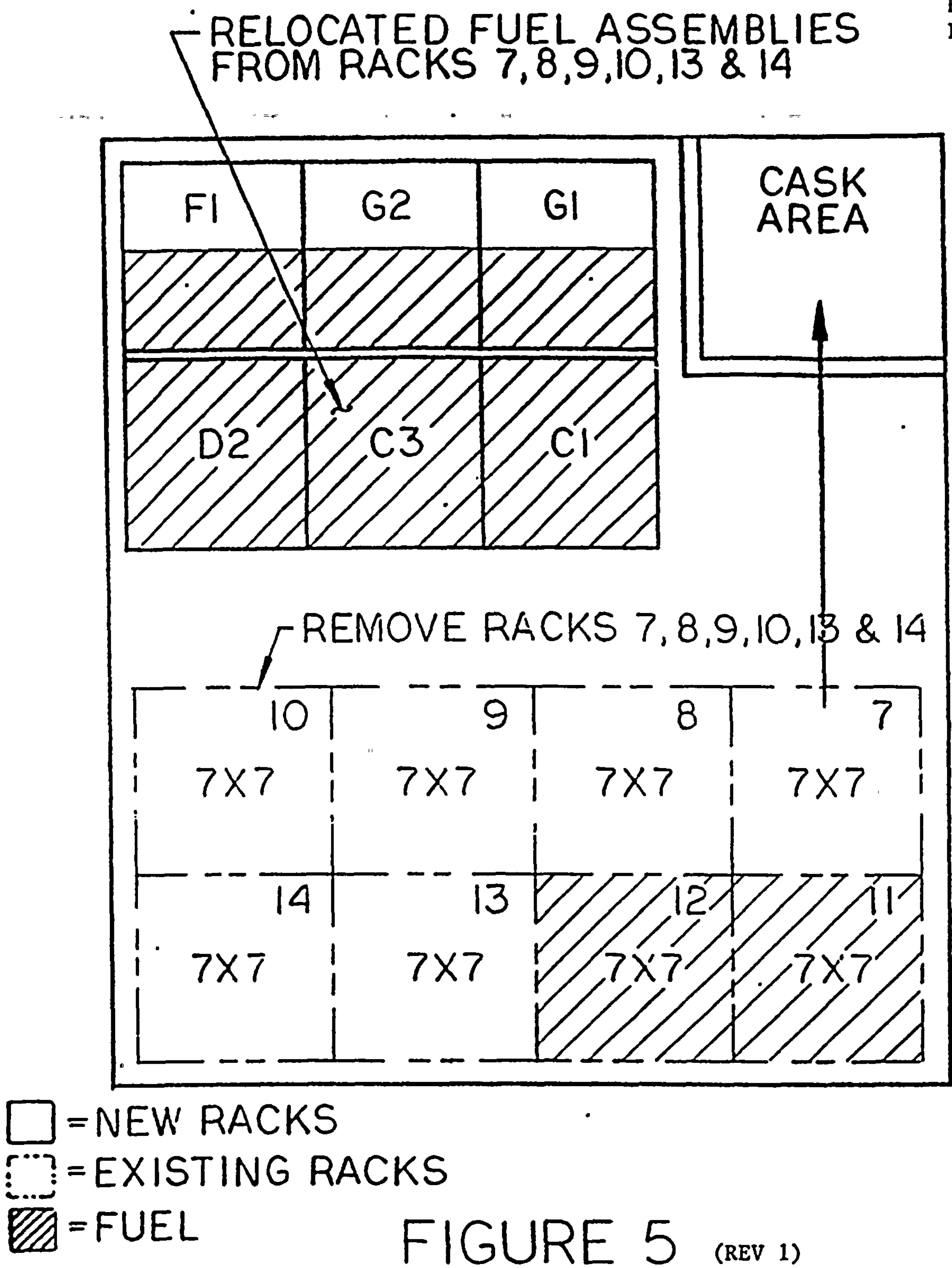

FIGURE 5 (REV 1)

NUC nuclear utility construction

TITLE: SEQUENCE OF RACK REMOVAL AND INSTALLATION

ENCLOSURE: 5 OF 8

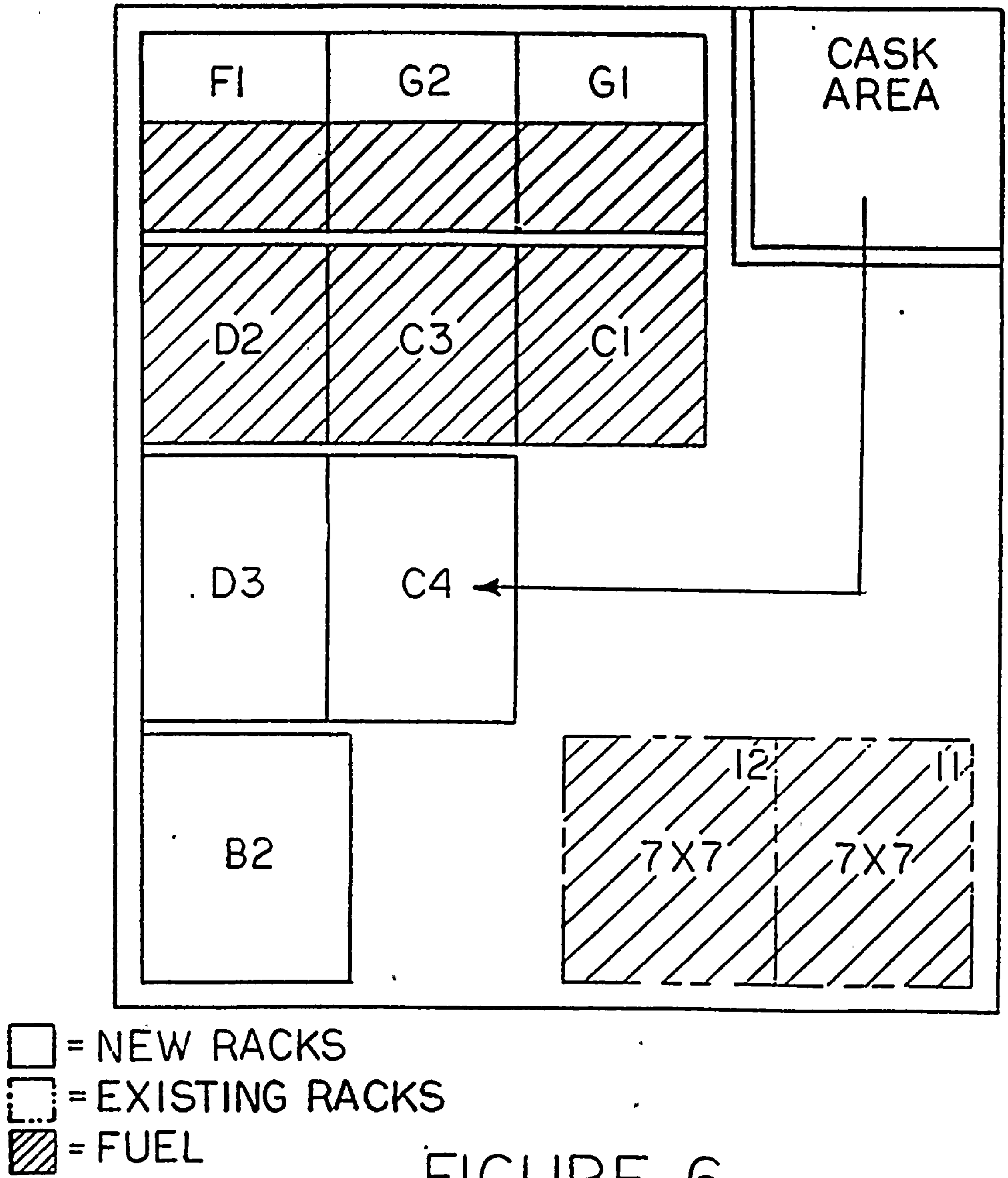

FIGURE 6 (REV 1)

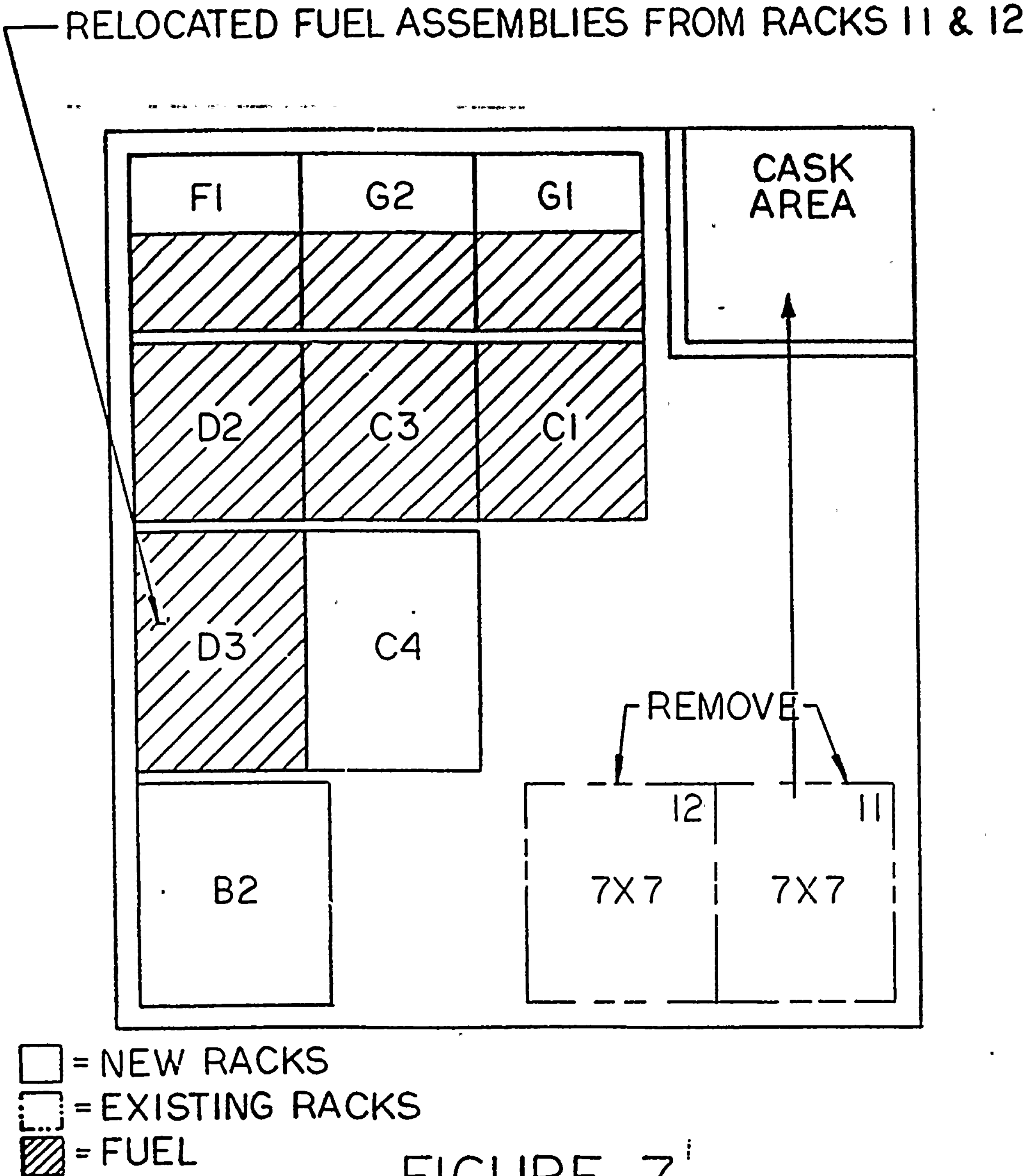

FIGURE 7

N

nuclear utility construction

TITLE: SEQUENCE OF RACK REMOVAL AND INSTALLATION

ENCLOSURE: 7 OF 8

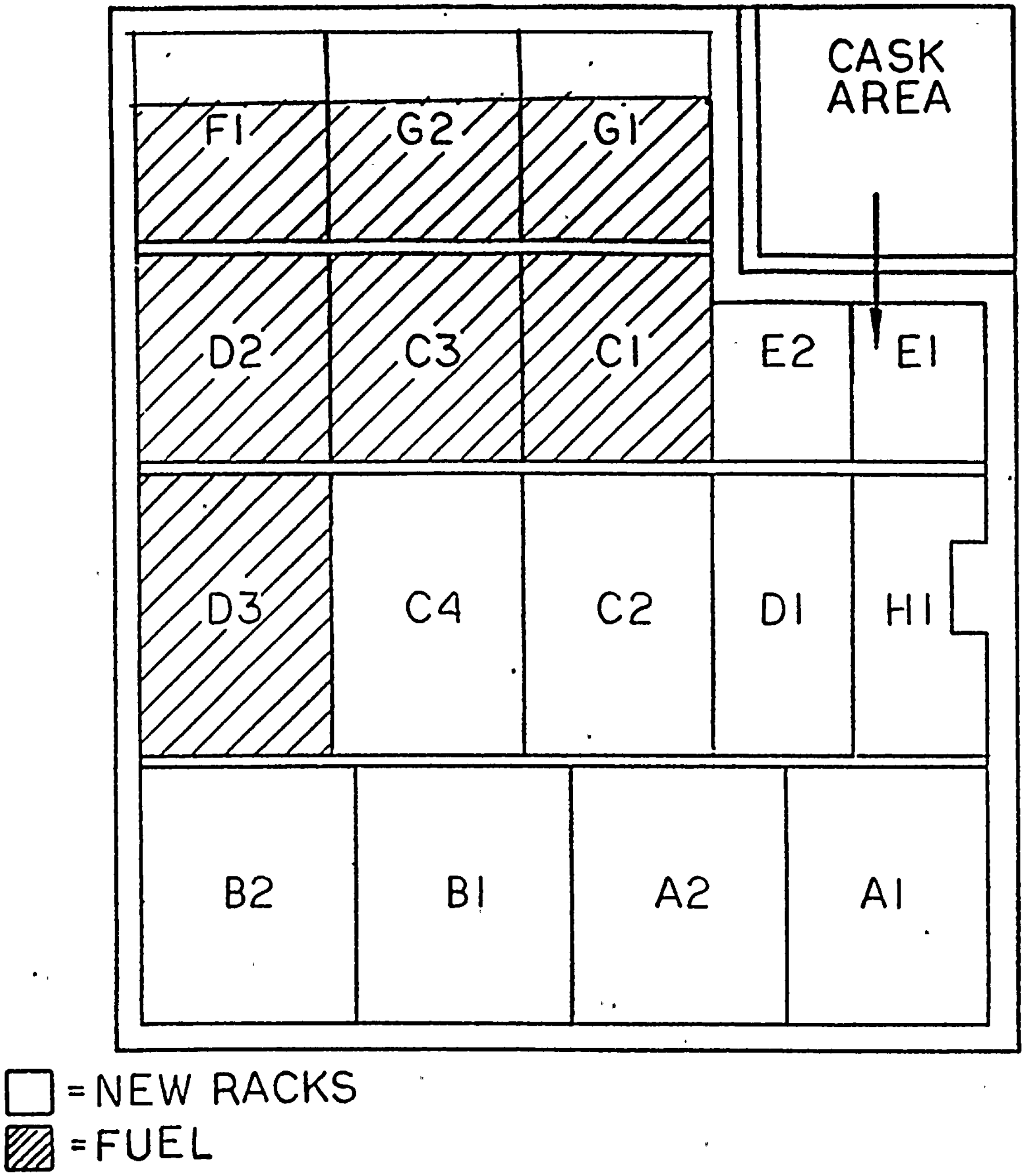
CASK
AREA
F1
G2
G1
D2
C3
C1
E2
E1
D3
C4
C2
D1
H1
B2
B1
A2
A1
=NEW RACKS
=FUEL

FIGURE 8


N

QUESTION #7 Provide additional discussion on the temporary crane that will be used to handle the spent fuel storage racks within the fuel building. As part of the discussion, specify 1) whether the crane is single failure proof; 2) whether the temporary crane will be carried over the spent fuel pool; and 3) the load path that the fuel building crane will take to carry the temporary crane. Provide drawings of the temporary crane and pertinent details which demonstrates the crane's compliance with the guidelines of NUREG-0612 and NUREG-0554. If the temporary crane is not single failure proof, provide the results of the analysis which demonstrates that the appropriate safety factors have been applied to the crane and that load handling by the crane will be in accordance with the guidelines of NUREG-0612.

RESPONSE: The temporary crane is of double girder, symmetrically mounted double hoist construction designed per the guidelines of the Crane Manufacturer's Association of America (Standard CMAA70). In addition, structural members in the crane structure meet the stress requirements of NUREG 0612. The crane traverses the pool in the north-south direction over rails which run adjacent to the pool edges resulting in an approximately 34'-6" crane span. The rack installation procedure calls for heavy loads to be restricted to the regions of the pool where there is no fuel stored in the racks. The crane is not single failure proof, and is not qualified to handle fuel. It may, however, be used to perform dummy gage testing on the installed fuel racks to ensure that they meet the required opening size.

The temporary crane will be used as a staging platform on which to set the old racks while the lifting hardware is changed out to enable the racks to be removed from the fuel handling building. The spacing between the centerlines of the girders is accordingly set at 74.83 inches to coincide with center-to-center spacing of the old rack support channels. The temporary crane will also be used to set the new rack modules at their designated locations in the pool after the new racks are brought into the cask area by the cask crane.

The cask crane will carry the temporary crane through the hatch door opening in five pieces; two end trucks, two girders and one hoist. The 4 foot area in the north end of the pool will be utilized for installing the crane. While the temporary crane will be carried over the spent fuel pool, it will not be carried over spent fuel.

RADIOLOGICAL CONSEQUENCES OF A SPENT FUEL POOL LOSS OF COOLING ACCIDENT

1. INTRODUCTION

In the course of reviewing the proposed modification of the spent fuel storage facility at the St. Lucie Plant Unit 1, the NRC has requested FP&L to provide the results of an analysis to show that no off-site dose limits will be exceeded were a loss of the spent fuel pool cooling system to result in the spent fuel pool boiling.

2. METHODOLOGY

2.1 Assumptions

The analysis of the radiological consequences of a loss of the spent fuel pool cooling system is bounded by the following assumptions:

1. the reactor has been shut down for seven days prior to the start of the accident (this is consistent with [1:3-28]*)
2. one full core has been discharged into the spent fuel pool
3. the dynamics of the pool boiling are characterized by the data in ref. [1:3-37]
4. the leak rate coefficient for iodine leaking from defective fuel rods is the same as for rods in the reactor seven days after shutdown [2:391] (although this is data for a BWR, it is the only actually measured leak rate data available and should be representative of light-water reactors)
5. the radionuclide core inventory is taken from ref. [3:Table 3.3]+
6. the pool water clean-up system will become inoperative when the pool cooling system fails
7. no credit is taken for the emergency filtration system in the fuel handling building HVAC system
8. the tritium concentration in the pool water is taken from ref. [1:5-14]

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*Numbers in square brackets are reference numbers, followed by page or table number

+Change of fuel vendor and use of higher burnup dictates the use of an updated radionuclide core inventory.

9. the leak rate coefficient for noble gases is five times that for iodines [2:399]
10. one percent of the fuel rods in the assemblies in the pool are defective
11. the atmospheric dilution factors (Xi/Q) at the nearest site boundary distance and at the outer boundary of the low population zone (LPZ) are taken from ref. [8:2.3-24]
12. the leakage of halogens and noble gases from spent fuel from previous refuelings is negligible
13. the concentrations of halogens and noble gases in the spent fuel pool at the start of the accident are negligible
14. all halogens* and noble gases leaking from the fuel rods will evolve directly into the atmosphere following pool boiling
15. the water level inside the fuel pool will be 23 feet above the top of the fuel at the start of the accident
16. sufficient make-up water will be provided to keep the fuel covered with water at all times
17. the accident is presumed to last 30 days+.

*See "Discussion", below.

+This assumption is consistent with analyses of other design basis accidents involving prolonged releases, such as a reactor loss-of-coolant accident (LOCA).

2.1.1 Discussion

The assumption that all halogens leaking from the fuel rods are released directly to the atmosphere is highly conservative. It is supported by the general principle that dissolved gases become less soluble with increasing temperatures, with the solubility becoming essentially zero at the boiling point of the solvent. Furthermore, since the cleanup system is assumed to have failed, there is no mechanism for removing the build-up of radioidines except evaporation. The concentration in the water will thus increase while the accumulation in the air will be removed by the ventilation system. Eventually, a quasi-static equilibrium will be achieved--the concentration in the water will reach a maximum value and the rate of volatilization from the pool will be equal to the rate of release from the fuel.

This simplified model leads us to conclude that, regardless of the actual partition coefficient, all the escaping iodines will eventually enter the atmosphere. A more detailed calculation of this non-equilibrium situation would result in somewhat lower release rates, and therefore lower radiological impacts. Such a calculation would take credit for non-volatile fractions of the radioiodine and also would spread the releases over a longer time period, resulting in lower concentrations due to radioactive decay. Performing such a calculation does not appear to be justified, however, since reliable data for the behavior of radioiodine in boiling pool water is not readily available.

2.1.2 Input data

<u>Atmospheric Dilution Factors (Xi/Q)</u>

Location	Time After Accident	Xi/Q (sec/m^3)
Exclusion area boundary	0 - 2 hrs	8.55E-5*
Low population zone	0 - 8 hrs	7.97E-6
	8 - 24 hrs	4.26E-6
	1 - 4 days	9.63E-7
	4 - 30 days	3.68E-7

*8.55E-5 = 3.55 x 10^{-5}

<u>Fission Product Core Inventory:</u>

Nuclide[a]	Half-life[b]	Core Inventory (Ci)	
		During operation	7 days after shutdown[c]
^{131}I	8.02 days	7.84E7	4.28E7
^{132}I	78.2 hours[d]	1.12E8	2.53E7
^{133}Xe	5.25 days	1.47E8	5.83E7
^{85}Kr	10.7 years	1.01E6	1.01E6

[a]Nuclides significant to the present analysis
[b]See ref. [7]
[c]Values calculated, based on equilibrium inventory and radioactive decay
[d]Half-life of parent tellurium-132 (^{132}Te)

The half-life of ^{132}I is 2.3 hours. However, it is in secular equilibrium with its parent nuclide, ^{132}Te. Therefore, the decrease of its activity in the core seven days after shutdown will be almost exactly the same as the decrease of the activity of its parent.

2.2 METHOD OF CALCULATION

The data in reference [2] indicate that the fractional leak rate of a given nuclide from the defective fuel rods seven days after shutdown would depend primarily on its chemical properties and secondarily on its half-life. That report also observes that the ratio of iodine to noble gas releases during operation was approximately 1:5 for nuclides with similar half-lives. The present analysis assumes that the leak rate coefficient of ^{132}I is the same as that given for ^{131}I (the data indicates that it would be somewhat smaller). The rates of ^{133}Xe and ^{85}Kr are calculated based on the data in that report.

The absorbed dose commitment to the adult thyroid for radioiodines at a given location is calculated as follows:

$$D_{ij} = \frac{I_i f r X_j B_j R_i (e^{-k_i t_{jo}} - e^{-k_i t_{j1}})}{k_i}$$

D_{ij} = adult absorbed dose due to nuclide i during time period j

I_i = core inventory of nuclide i at commencement of pool boiling

f = fractional leak rate of damaged fuel rod
= 8.7E-10 sec^{-1} (^{131}I)

r = fraction of failed fuel
= .01 (1%)

X_j = Atmospheric dilution factor (Xi/Q) during time period j from start of accident (sec/m^3)

B_j = adult breathing rate during time period j [4:1.4-2]
= 1.25 m^3/hr (0 - 8 hrs from start of accident)
= .625 m^3/hr (8 - 24 hrs ")
= .833 m^3/hr (1 - 30 days ")

R_i = Adult dose conversion factor (rads/curie inhaled) [5:25.5]
= 1.48E6 for ^{131}I
= 5.35E4 for ^{132}I

k_i = decay constant of nuclide i (hr^{-1})

t_{jo} = time from start of accident to beginning of period j

t_{j1} = time from start of accident to end of period j

The doses to the whole body and to the skin from noble gases are calculated as follows:

$$D_{ij} = \frac{I_i f r X_j R_i (e^{-k_i t_{jo}} - e^{-k_i t_{j1}})}{k_i}$$

All terms have the same meaning as in the previous equation, except:

R_i = semi-infinite cloud dose conversion factor [6:1.109-21]
f = 4.35E-9 sec^{-1} (^{133}Xe)

The atmospheric tritium concentration is calculated as follows:

$C = FcX_j$

C = tritium concentration in air (uCi/m^3)
F = boil-off rate of pool water
= 4,377 ml/sec
c = tritium concentration in pool water
= .077 uCi/ml

3. RESULTS

3.1 Dose Commitment at Exclusion Area Boundary (EAB)

Nuclide	Adult absorbed dose commitment at EAB (rem)		
	Thyroid	Skin	Whole Body
^{131}I	0.117	---	---
^{132}I	2.50E-3	---	---
^{133}Xe	---	1.51E-5	1.45E-5
^{85}Kr	---	1.15E-6	1.38E-8
Total	0.119	1.62E-5	1.45E-5

3.1.1 Tritium concentration at EAB

During the first two hours following the accident, the maximum tritium concentration at the Exclusion Area Boundary (EAB) is calculated to be 2.9E-8 uCi/ml (2.9E-2 uCi/m^3) in air. This value is well below the maximum permissible concentration (MPC) for tritium, which is 2E-7 uCi/ml. The MPC is the limiting concentration for a chronic exposure. Since an individual at the EAB is postulated to be exposed for a maximum of two hours, the contribution of tritium to his total dose is insignificant.

3.2 Dose Commitment at LPZ

The following calculations were performed for the maximum exposed individual at the LPZ outer boundary during the 30-day period of the accident.

Nuclide	Adult absorbed dose commitment at LPZ (rem)		
	Thyroid	Skin	Whole Body
^{131}I	0.121	---	---
^{132}I	1.92E-3	---	---
^{133}Xe	---	1.89E-5	1.82E-5
^{85}Kr	---	2.89E-6	3.48E-8
Total	0.123	2.18E-5	1.82E-5

3.2.1 Tritium concentration at LPZ

During the first eight hours following the accident, the maximum time-averaged tritium concentration at the LPZ boundary is calculated to be 2.7E-9 uCi/ml in air. This value is well below the MPC for tritium. Since the atmospheric dilution factor (Xi/Q) used in this calculation is applicable to the first two hours of the accident only, and since in any case the accident is postulated to last only 30 days, the contribution of tritium to the total dose is insignificant.

4. CONCLUSIONS

The maximum calculated thyroid dose is much less than 1% of the 10 CFR 100 limit of 300 rem. The whole-body and skin doses are completely insignificant.

REFERENCES

1. Florida Power & Light Company: "St. Lucie Plant - Unit No. 1, Spent Fuel Storage Facility Modification Safety Analysis Report, Docket No. 50-335.
2. Eickelpasch, N., and R. Hock: "Fission Product Release after Reactor Shutdown", IAEA-SM-178/19, in Experience from Operating and Fuelling Nuclear Power Plants, Vienna, 1974
3. Exxon Nuclear Company, Inc.: "St. Lucie Unit 1 Radiological Assessment of Postulated Accidents". XN-NF-84-85(P), Rev. 1.
4. U.S. Atomic Energy Commission, Regulatory Guide 1.4: "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors". Rev. 2, 1974.
5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.25: "Assumptions used for Evaluating the Potential Radiological Consequences of a Fuel

Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors".

6 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.109: "Calculations of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I". Rev. 1, 1977.

7. Lederer, C. M., and V. S. Shirley, eds: <u>Table of Isotopes</u>, 7th Ed., 1978

8. Florida Power & Light Company: "St. Lucie Plant - Unit No. 1, Updated Final Safety Analysis Report", Amendment No. 6.