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March 29, 1984
RBG- 17458
File Code No. G9.5, G9.8.6.2

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

Dear Mr. Denton:

River Bend Station - Unit 1
Docket No. 50-458

Enclosed for your review are Gulf States Utilities Company responses to the open items identified in the Draft Safety Evaluation Report by the Auxiliary Systems Branch and responses to the request for additional information identified in part by Staff letters dated August 5, 1981 and December 31, 1981. This letter supplements docketed correspondence from Mr. Booker to Mr. Denton dated December 1, 1983, December 30, 1983, February 2, 1984 and March 6, 1984. Attachment 1 summarizes the open items and indicates changes to be made in the River Bend Station FSAR. Attachment 2 provides a brief discussion of each open item, the response and reference material for each item. Where indicated, these responses will be provided in a future amendment to the FSAR.

Sincerley,

Eddie R Grant

for J. E. Booker
Manager-Engineering,
Nuclear Fuels & Licensing
River Bend Nuclear Group

JEB/WJR/ERG/JEP/je

Enclosures

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Attachment 1

ITEM NUMBER	DSER SECTION	SUBJECT	FSAR REVISIONS
8.	9.1.2 Pg. 9-10,11	Spent Fuel Racks Criticality Analysis	Enclosure 1 Q410.6
10c.	9.1.4 Pg. 9-27	Light Load Maximum Kinetic Energy	Enclosure 2 Q410.36

Attachment 2

8. The applicant has not provided a criticality analysis to confirm the criticality limits to be attained in the spent fuel storage facility. (DSER Pages 9-10 & 11).

RESPONSE

A summary of the criticality analysis was provided with our letter of February 2, 1984. This summary is reflected in the FSAR revisions provided in Enclosure 1 which supplements responses submitted March 6, 1984.

10c. The applicant should provide an analysis to show that the maximum kinetic energy resulting from a fall of any object which weighs less than fuel bundle and its handling tool, when over spent fuel in either containment or the fuel building storage facilities will not exceed that obtained in the fall of a fuel bundle and its handling tool (fuel handling accident). (DSER Pages 9-24, 26 & 27.)

RESPONSE

Tables 1, 2, and 3 of Enclosure 2 have been prepared in response to this question. The tables provide a listing of items that may fall onto the racks containing spent fuel in either the containment or fuel buildings, or into the reactor core itself during the refueling operation. The tabulations show the following:

a) In the fuel building, the kinetic energy of these items does not exceed that of a channeled fuel assembly as described in Section 15.7.4.

b) An evaluation of the consequences of dropping any of the objects listed in Tables 2 and 3 onto spent fuel in the dryer storage pool and reactor core, respectively, verifies that the conclusions stated in FSAR Section 15.7.4.1.1 remain valid.

ENCLOSURE 1

QUESTION 410.6 (9.1.2.3.1.2)

Provide a schedule for submittal of the criticality analysis for the fuel building spent fuel storage or provide the analysis.

RESPONSE

The criticality analysis for the fuel building spent fuel storage racks ~~will be provided by December 1983~~ is provided in Section 9.1.2.3.1.2. 7

3. Fuel stored in control rod guide tube racks (Fig. 9.1-5)

4. Pool water temperature increases to 212°F

5. Two bundles placed side by side while separated from the storage rack area by 12 in of water (Fig. 9.1-6)

6. Three-bundle linear array separated from the storage rack area by 12 in of water (Fig. 9.1-6)

7. Three-bundle tee array separated from the storage rack area by 12 in of water (Fig. 9.1-6)

8. Normal storage array of ruptured fuel

9. Abnormal condition of pool being drained and ruptured fuel containers being flooded

10. Moving fuel bundle between work rack and storage area

11. Moving fuel bundle in aisle between storage racks

12. Grapple drop displacing two fuel bundles (Fig. 9.1-6)

13. Four-bundle square array separated from the storage rack area by 12 in of water (Fig. 9.1-6).

Concerning safety implications related to sharing, no limitation is placed on the size of the spent fuel storage array from a criticality standpoint, since all calculations are performed on an infinite basis.

9.1.2.3.1.2 Fuel Building Fuel Storage

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~~The fuel building fuel pool storage design incorporates poison-type, high-density spent fuel storage racks. Details of the criticality analysis will be provided in a future amendment.~~ INSERT

9.1.2.3.2 Structural Design

7 9.1.2.3.2.1 Containment Fuel Storage (Refer to Figs. 9.1-2 and 9.1-2a)

1. The spent fuel pool in the containment building contains 20 sets of racks which may contain up to

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Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between assemblies. Structural design of the racks is discussed in Section 9.1.2.3.2.2.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (K_{eff}) of the fuel assembly array is less than 0.95 as recommended in ANSI N210-1976[4] and in an NRC staff position letter.[1]

In meeting this design basis, some of the conditions assumed are: General Electric 8 x 8 BWR/6 fuel with an enrichment of 3.80 w/o U-235 are stored, the pool water has a density of 1.0 gm/cm^3, the storage arrray is infinite in lateral and axial extent which is more reactive than the actual finite array, mechanical and method biases and uncertainties are included, the minimum poison loading is used, and no credit is taken for any burnable poison in the fuel assemblies.

The design method which determines the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX system [5] of codes for cross-section generation and the KENO IV Code [6] for reactivity determination. A set of 27 critical experiments [7,8,9] has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and variability which are then included in reactivity analysis of the rack.

The result of the above considerations is that the nuclear design of the River Bend Station racks meet the requirements of its specification as well as U.S. Nuclear Regulatory Commission guidelines [1] and criteria.[10]

INSERT - For Page 9.1-10

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between assemblies. The spent fuel rack is designed to be in compliance with General Design Criteria 62.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (keff) of the fuel assembly array will be less than 0.95 as recommended in ANSI N210-1976 (Ref. 4) and in Reference 1. This design basis applies for both normal and abnormal fuel storage conditions. Normal conditions exist when the spent fuel storage racks are located in the pool and are covered with a normal depth of water. An abnormal condition may result from accidental dropping of a fuel assembly or equipment, or loss of spent fuel pool cooling.

The design method which determines the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX system (Ref. 5) of codes for cross-section generation and the KENO IV Code (Ref. 6) for reactivity determination. A set of 27 critical experiments (Ref. 7,8,9) has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and variability which are then included in reactivity analysis of the rack.

For normal fuel storage, a model is developed based on the following assumptions:

1. The fuel stored is General Electric 8 x 8 BWR/6 fuel. See Table 4.2-4 for fuel assembly parameters. The fuel channels are included in the model, which is conservative.

2. All fuel rods contain uranium dioxide at an enrichment of 3.80 w/o U-235 over the infinite length of each rod, and no credit is taken for any of the gadolinium poison in the fuel rods.

3. No credit is taken for any U-234 or U-236 in the fuel, nor is any credit taken for the buildup of fission product poison material.

4. The moderator is pure water at the temperature within the design limits of the pool which yields the largest reactivity. A conservative value of 1.0 gm/cm3 is used for the density of water.

5. No credit is taken for any fuel assembly spacers or Inconel springs.

6. The minimum poison material loading (i.e. 0.02 grams B-10 per square centimeter) and minimum poison thickness (0.068 in.) are used throughout the array.

7. The array is infinite in lateral and axial extent which precludes any neutron leakage from the array, and is thus more reactive than the actual finite array. Figure 9.1-3 (sheet 4 of 4) shows the nominal dimensions of individual cell modules and indicates the unit cell modeled in the "infinite array".

The calculation of Keff for the above nominal case results in a Keff of 0.9190 with a 95 percent probability/95 percent confidence level uncertainty of + 0.00379. When mechanical and method biases and uncertainties are included (e.g., tolerances for mechanical construction that can result in reduced cell center-to-center spacings, and storage cell material thickness), the total uncertainty is 0.02 Delta K resulting in a Keff of 0.9390; thus, the acceptance criterion for criticality is met.

The following abnormal fuel storage conditions were analyzed.

1. Drop of a fuel assembly on top of the storage racks.

2. Drop of a fuel assembly next to the unpoisoned periphery of the racks.

3. Drop of an object which imposes an impact energy of 3800 ft-lbs on the rack.

4. Loss of spent fuel pool cooling system.

5. Water injection into the racks when used for dry storage.

For all of the above abnormal conditions, the criticality analysis shows that the acceptance criterion for criticality is met.

TABLE 4.2-4

FUEL DATA

Core

Number of fuel assemblies	624
Fuel cell spacing (control rod pitch), in	12.0
Total number of fueled rods[1]	38,688
Core power density (rated power) kW/l	52.4
Total core heat transfer area, ft^2	61,152

Fuel Assembly Data

Nominal active fuel length, in[2]	150
Fuel rod pitch, in	0.636
Fuel rod spacing, in	0.153
Fuel bundle heat transfer area, ft^2	98
Fuel channel wall thickness, in	0.120
Channel width (inside), in	5.215
Fuel Channel Material	Zircaloy-4

Fuel Rod Data

Outside diameter, in	0.483
Cladding inside diameter, in	0.419
Cladding thickness, in	0.032
Fission gas plenum length, in	10
Pellet immersion density, % theoretical density	95
Pellet outside diameter, in	0.410
Pellet length, in	0.410
Fuel Rod Clad Material	Zircaloy-2

Water Rod Data

Outside diameter, in	0.591
Inside diameter, in	0.531
Water Rod Material	Zircaloy-2

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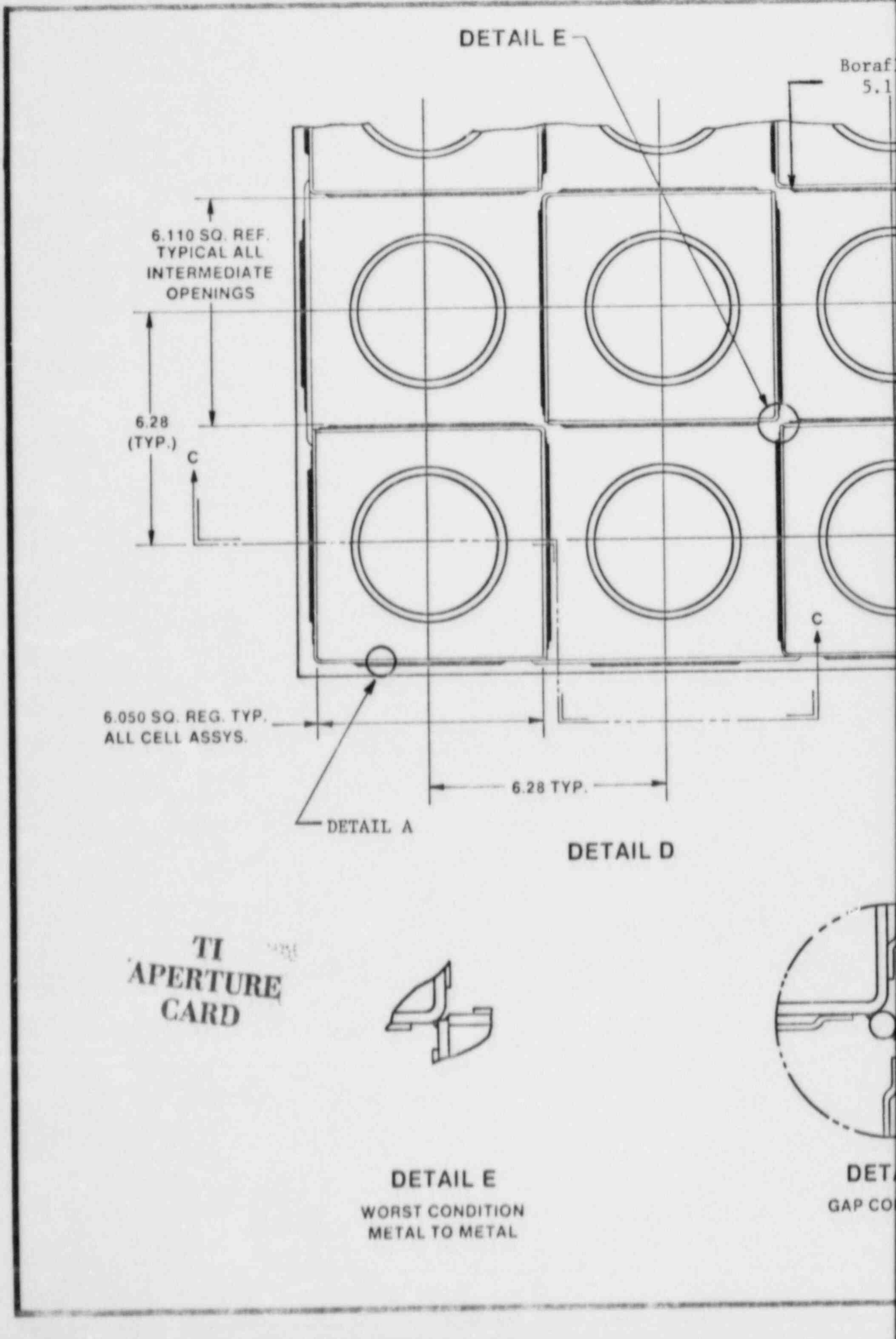
[1] Does not include two water rods in each assembly
[2] Includes six in of natural U at the top and bottom of the fuel column



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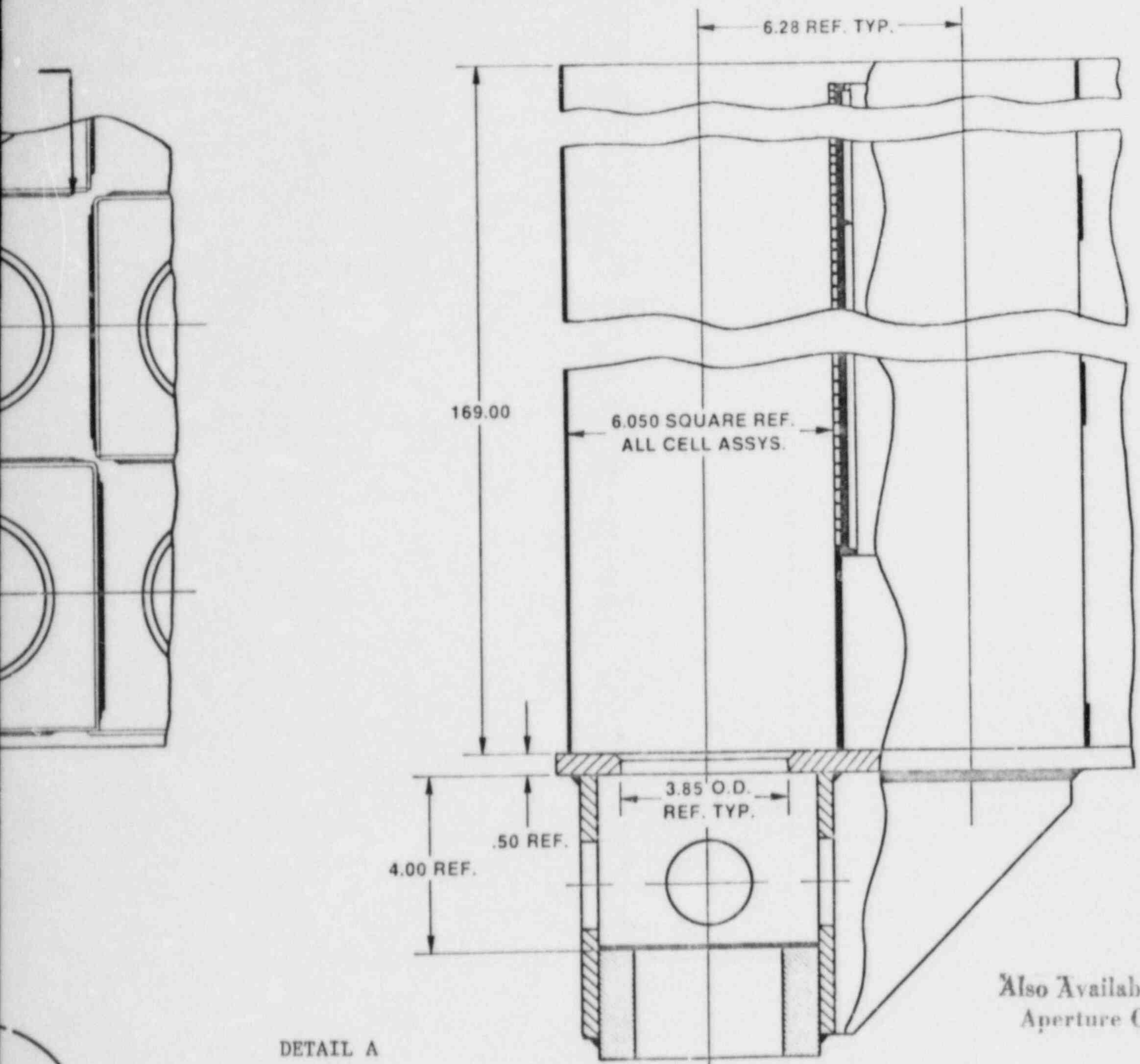
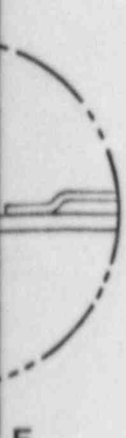
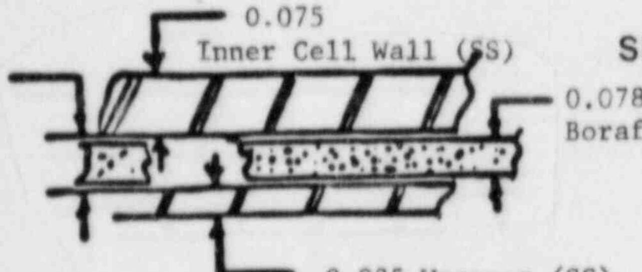


DETAIL A





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SECTION C-C

FIGURE 9.1-3

SPENT FUEL STORAGE — FUEL BUILDING
(SHEET 4 OF 4)

RIVER BEND STATION
FINAL SAFETY ANALYSIS REPORT

ENCLOSURE 2

TABLE 1

	Item	Distance (feet) Above Pool Surface	Total Above Rack	Dry Weight (lbs)$^{(2)}$	Kinetic Energy at Impact (top of rack (ft.-lbs)
1.	Channel bolt wrench	4	31	15	465
2.	Channel handling tool	11.5$^{(1)}$	38.5$^{(1)}$	25	963
3.	Channel gauging fixture		17.8$^{(1)}$	210	3738
4.	General purpose grapple	3.6$^{(1)}$	30.6$^{(1)}$	45	1377
5.	Actuating pole	4	31	61	1891
6.	Drop Light	4	31	25	775
7.	Local area underwater light	4	31	35	1085
8.	Viewing aid		27	13	351
9.	Control rod		12$^{(1)}$	218	2616
10.	Hand rail, removable	4	31	122	3782
11.	Underwater TV camera	4	31	50	1550
12.	Defect. Fuel Contain w/fuel assy.		5.4	573	3094
13.	Fuel channel		24$^{(1)}$	99	2376
14.	Dummy fuel assembly		6.1	621	3803$^{(3)}$
15.	Fuel bundle		6.1	538	3296$^{(3)}$
16.	Spent Fuel Assembly		6.1	621	3803$^{(3)}$

Notes:

(1) Distance from CG to top of fuel storage rack and pool surface.
(2) Kinetic energy calculated without credit taken for buoyancy and drag for conservatism, except for defective fuel container & spent fuel assembly.
(3) See FSAR Section 15.7.4 for fuel assembly fuel handling accident. Kinetic energy value differs because of rounding-off of factors.

TABLE 2

Light Loads Over The Dryer Storage Pool (Containment)

Item	Distance Above HWL (ft.)	Total Distance Above Rack (ft)	Dry Weight (lbs)	Kinetic Energy at Impact (top of rack (ft.-lbs)(2)
Channel Bolt Wrench	16'-2"	41'-9"	15	626
Channel Gaging Fixture	0	15'-7"	210	3272
General Purpose Grapple	1'-8"	27'-3"	45	1226
Utility Manipulator	19'-8"	45'-3"	45	2036
Actuating Pole	0	15'-0"	61	915
Fuel Bundle Sampler	0	12'-0"	600	7200
Drop Light	4'-2"	29'-9"	25	744
Spent Fuel Assembly	0	4'-10"	621	3012
Viewing Aids	0	25'-7"	13	333
Light Support Bracket	4'-2"	29'-9"	110	3273
Channel Handling Tool	12'-2"	37'-9"	25	944
Underwater Viewing Tube	4'-2"	29'-9"	20	595
Fuel Support Grapple	19'-2"	44'-9"	147	6578
Peripheral Fuel Support Plug	20'-2"	45'-9"	16	732
Control Tube Grapple	16'-2"	41'-9"	45	1879
Control Rod Latch Tool	0	21'-9"	74	1610
Blade Guide	0	21'-9"	180	3915
Grid Guide	0	21'-9"	32	696
Removable Handrails	2'-2"	27'-9"	122	3386
Magnetic Retriever	4'-2"	29'-9"	2	60

TABLE 3

Light Loads Over The Reactor Vessel Core

Item	Distance Above HWL (ft.)	Total Distance Above Core (ft)	Dry Weight (lbs)	Kinetic Energy at Impact (top of rack (ft.-lbs)
General Purpose Grapple	17'-2"	71'-7"	45	3221
Utility Manipulator Grapple	20'-2"	74'-7"	45	3356
Fuel Bundle Sampler	0	40'-10"	600	24500
Clam Shell Retriever	3'-0"	57'-5"	25	1435
Peripheral Fuel Support Plug	20'-2"	74'-7"	16	1193
Drop Light	20'-2"	74'-7"	25	1865
Underwater TV	18'-8"	73'-1"	25	1827
Viewing Aid	0	54'-5"	13	707
Spent Fuel Assembly	0	34'-0"	621	21107
Fuel Support Grapple	19'-2"	73'-7"	147	10817
Peripheral Office Grapple	17'-2"	71'-7"	45	3221
CRD Guide Tube Seal	20'-2"	74'-7"	60	4475
In-Core Guide Tube Seal	0	52'-7"	56	2945
Peripheral Orifice Holder	20'-2"	74'-7"	130	9696
Blade Guide	0	52'-1"	180	9375
Fuel Bail Cleaner	15'-0"	69'-5"	100	6917
Grid Guide	0	51'-5"	32	1645
Fuel Grapple	9'-2"	63'-7"	450	28613
Control Rod Grapple	19'-2"	73'-7"	17	1251
CRD Guide Tube Grapple	19'-8"	74'-1"	35	2593
Stud Handling Tool	17'-2"	71'-7"	135	9664

Table 3, Continued

Light Loads Over The Reactor Vessel Core

Item	Distance Above HWL (ft.)	Total Distance Above Core (ft)	Dry Weight (lbs)	Kinetic Energy at Impact (top of rack (ft.-lbs)
RPV Stud	18'-2"	72'-7"	500	36292
Shroud Mead Bolt Wrench	18'-2"	72'-7"	40	3023
Steamline Plug Installation Tool	13'-2"	67'-7"	72	4866
Local Area Underwater Light	20'-2"	74'-7"	35	2610
General Area Light	20'-2"	74'-7"	50	3729
Steamline Plug	19-2"	73'-7"	375	27594
Head Stud Rack	18'-2"	72'-7"	200 (empty)	14517
Magnetic Retriever	5'-0"	59'-5"	2	119

Notes: (For Tables 2 and 3)

(1) All distances are measured from the C.G. of the dropped object to the top of either the rack or core.

(2) For conservatism, all kinetic energies are calculated without credit taken for buoyancy and drag forces, for the distance travelled thru water. The kinetic energies are therefore calculated as if the entire drop occurs thru air only. (Buoyancy effects were considered for the spent fuel assembly)