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January 24, 1979

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

Subject: Zion Station Units 1 and 2
Additional Information on Proposed
Expansion of Spent Fuel Storage Capacity
NRC Docket Nos. 50-295 and 50-304

Dear Mr. Denton:

The NRC Staff requested Commonwealth Edison Company to provide additional information in support of its request to expand the storage capacity of the Zion Units 1 and 2 spent fuel pool. The request consisted of two sets of questions telecopied from the Staff on November 14 and 28, 1978. Attachments 1 and 2 to this letter contain Commonwealth Edison's responses to these questions.

Please address any additional questions that you might have to this office.

One (1) signed original and thirty-nine (39) copies of this letter are provided for your use.

Very truly yours,

William F. Naughton

William F. Naughton
Nuclear Licensing Administrator
Pressurized Water Reactors

attachments

7902020234

ATTACHMENT 1

SPENT FUEL POOL CAPACITY EXPANSION

ZION NUCLEAR POWER PLANTS, UNITS 1 AND 2

DOCKET NOS. 50-295 AND 50-304

ROUND 2 QUESTIONS

1. QUESTION

In regard to your response number 20, a limit on the fuel assembly loading is more inclusive than a limit on the enrichment. Also, this maximum fuel loading can be obtained from just an arithmetical calculation of quality assurance data. For these reasons, we find that a technical specification on these racks which limits the fuel loading to 39.4 grams of Uranium-235, or equivalent, per axial centimeter of fuel assembly is an acceptable method of limiting the uncertainty in k_{eff} whereas a limit on the enrichment is not.

ANSWER

The appropriate technical specification change is being drafted and will be submitted after approval by our On-Site and Off-Site Review functions.

ATTACHMENT 1

2. QUESTION

In regard to your response number 21, state the bases for the dimensions of the cylindrical supercell (Figure 3) for the first benchmark calculation.

ANSWER

Radii R_1 and R_2 of the cylindrical supercell (Figure 3) is obtained by conserving the corresponding areas of the 9 x 9 basic fuel pin assembly. Radius R_3 is obtained by adding the aluminum wall thickness to the fuel pin assembly and then conserving the area. Radius R_4 is obtained by adding to R_3 half the thickness of the boral core.

Constants used for cylindrical supercell dimensions:

Basic fuel pin assembly array	=	9 x 9
Fuel rod pitch	=	0.75 inches
Thickness of AL wall	=	.041 inch
Thickness of boral plate	=	0.168 inch
Thickness of water layer	=	½(0.75-2x0.041-0.168)
	=	0.25 inch

Area of fuel region:

$$= (\text{fuel rod pitch} \times 9)^2$$
$$= (0.75 \times 9)^2 \text{ sq. in.}$$
$$= (6.75)^2 \text{ sq. in.}$$
$$= (6.75 \times 2.54)^2 \text{ sq. cm.}$$
$$= 293.95102 \text{ sq. cm.}$$
$$= \pi R_1^2$$
$$R_1 = \left(\frac{293.95102}{\pi}\right)^{1/2} \text{ cm.}$$
$$= 9.67303 \text{ cm.}$$

2. ANSWER (continued)

Area of (H_2O + fuel) region

$= \pi R_2{}^2$

$= (6.75 + \text{thickness of water layer})^2$ sq. in.

$= (6.75 + 0.25)^2$ sq. in.

$= (7 \times 2.54)^2$ sq. cm.

$= 316.1284$ sq. cm.

$$R_2 = \left(\frac{316.1284}{\pi}\right)^{1/2}$$

$= 10.03129$ cm.

Area of (H_2O + Aℓ + fuel) region

$= (7 + \text{thickness of AL layer})^2$ sq. in.

$= (7.041 \times 2.54)^2$ sq. cm.

$= 319.84246$ sq. cm.

$= \pi R_3{}^2$

$$R_3 = \left(\frac{319.84246}{\pi}\right)^{1/2} \text{ cm.}$$

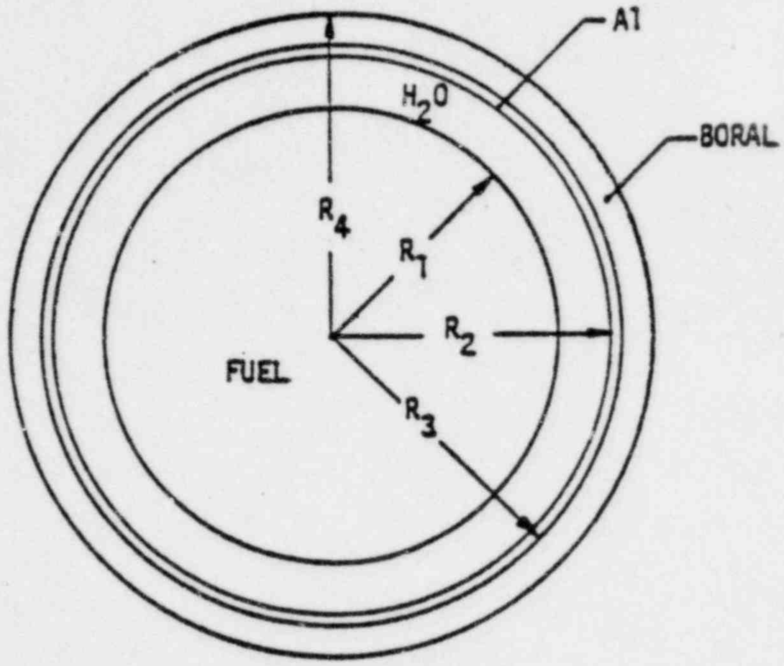
$= 10.09005$ cm.

$R_4 = R_3 + \frac{1}{2}$ thickness of boral plate

$= 10.09005 + \frac{1}{2} \times 0.168 \times 2.54$

$= 10.30341$ cm.

ATTACHMENT 1



R_1 = 9.67303 CM
R_2 = 10.03129 CM
R_3 = 10.09005 CM
R_4 = 10.30341 CM

Figure 3. 1-D Supercell Configuration for First Benchmark Calculations

ATTACHMENT 1

3. QUESTION

In your calculations of the two critical assemblies with Boral, describe how you accounted for the self-shielding of the boron carbide particles in the aluminum matrix.

ANSWER

In the benchmark calculations, the Boral core was homogenized and then the cross sections were obtained from XSDRN, which is a one dimensional discrete ordinates spectral averaging code. There was no account for particle self-shielding since the range of particle size is 60-200 mesh with a mean size of 175 mesh and, therefore, self-shielding effects are negligible.

ATTACHMENT 1

4. QUESTION

In regard to your response Number 23, the NRC requires an on-site neutron attenuation test to verify the presence of the boron. This is in addition to the Quality Assurance Program you described. Provide a description of the neutron attenuation test that you will perform at the Zion plant to statistically show with 95 percent confidence that the boron is not missing from more than one out of every sixteen plates.

ANSWER

A neutron posion verification test will be conducted at the Zion plant after the racks are installed in the pool. This will be a qualitative test to statistically show with 95 percent confidence that the boron is not missing from more than one out of every sixteen plates.

This procedure is similar to the poison verification tests conducted at Montecello and TVA by National Nuclear Corporation utilizing their proprietary equipment.

ATTACHMENT 1

5. QUESTION

For the proposed type of racks, a surveillance program is required to show the continued presence of boron throughout the life of the racks. Provide a description of the boron surveillance program that you will perform.

ANSWER

See attachment "A", Neutron Absorber Sampling Plan - In Pool.

ATTACHMENT 1

ATTACHMENT "A"

NEUTRON ABSORBER SAMPLING PLAN - IN POOL

A sampling plan to verify the ability of a neutron absorber material employed in the high density fuel racks to withstand the long-term environment is described.

The test conditions represent a restricted flow of water over the neutron absorber material. The samples will be supported adjacent to and suspended from the fuel racks. Eighteen (18) test samples are to be fabricated in accordance with Figure 1 and installed in the pool when the racks are installed.

The procedure for fabrication and testing of samples shall be as follows:

1. Samples shall be cut to size and dried in an oven for five hours at 175°F, followed by a cycle at 600°F for three hours.
2. Samples shall be weighed immediately following removal from the oven and weight in milligrams recorded for each sample.
3. Samples shall be fabricated in accordance with Figure 1 and installed in pool.
4. Two samples shall be removed per schedule shown in Table 1.
5. Carefully cut samples apart at the weld without damaging the neutron absorber. Wash with a soft brush in a mild abrasive and detergent solution, immerse in nitric acid to remove surface products, followed by a rinse of clean water and alcohol. Dry in a 175°F oven for five hours, followed by a cycle at 600°F for three hours.
6. Weigh the samples and evaluate the weight change in the neutron absorber material in milligrams per square centimeter per year.

ATTACHMENT 1

7. If pitting is present, the depth of the four major pits are to be recorded and the average pit penetration in mils of an inch per year determined.

8. Retain two (2) samples.

9. Prepare report of sample test results and observations.

ATTACHMENT 1

TABLE 1

Date Installed __________

SAMPLE NO.	SCHEDULE	INITIAL WEIGHT (mg/Cm^2-Yr)	FINAL WEIGHT (mg/Cm^2-Yr)	WEIGHT CHANGE (mg/Cm^2-Yr)	PIT PENETRATION mil/Yr
1					
2	90 day				
3					
4	180 day				
5					
6	1 year				
7					
8	5 year				
9					
10	10 year				
11					
12	15 year				
13					
14	20 year				
15					
16	30 year				
17					
18	40 year				

ATTACHMENT 1

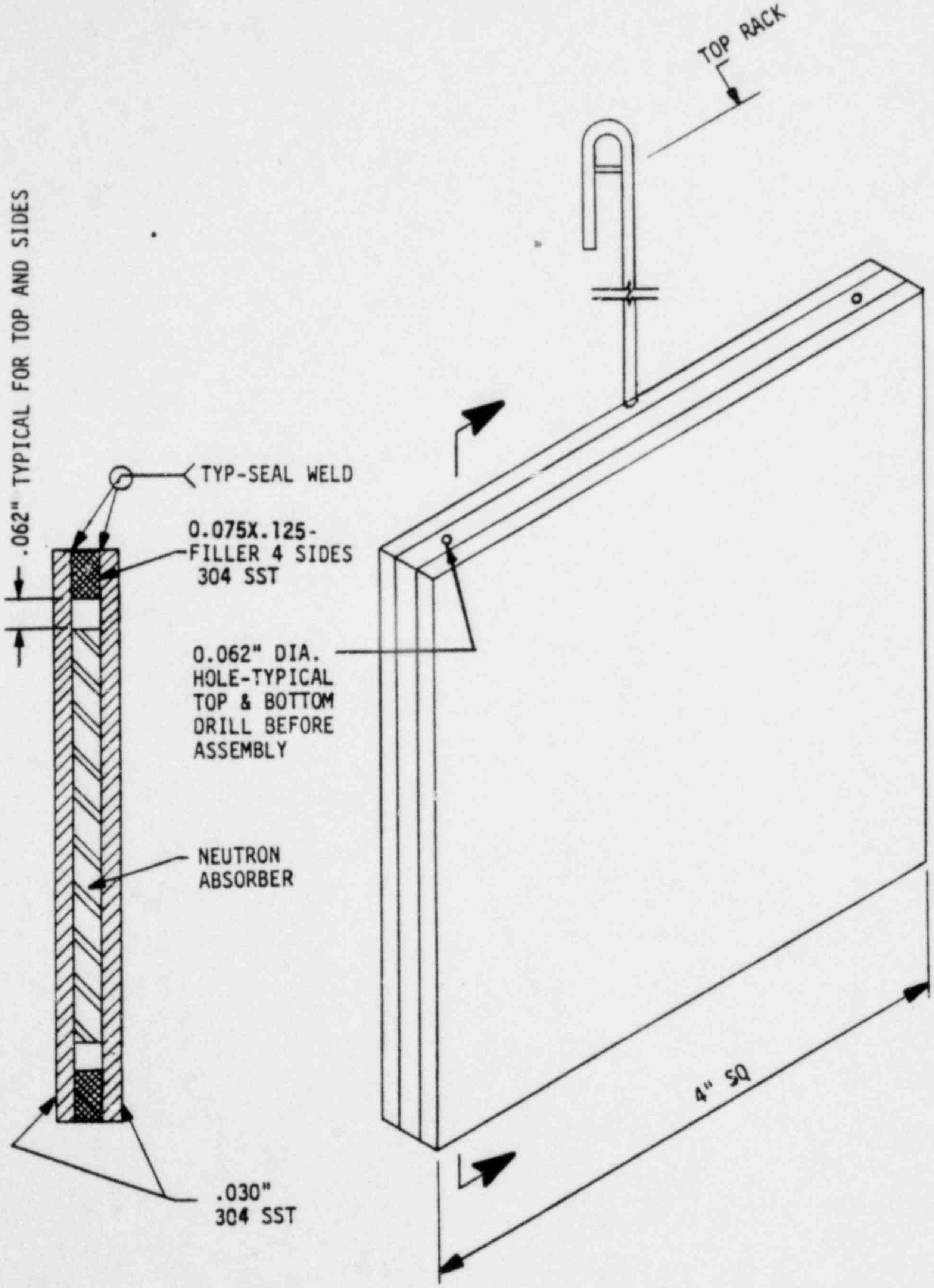


Figure 1

ATTACHMENT 2

SPENT FUEL POOL CAPACITY EXPANSION

ZION NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-295 AND 50-304

ROUND 3 QUESTIONS

QUESTION NUMBER 1:

Provide a more detailed description of the inter-tube welded connection; include drawings if possible. Specifically discuss if the tubes are welded continuously to each other the full length of the tube or only at discrete intervals. Also discuss the structural members or plates used in this connection.

RESPONSE:

The tubes typically have bars (flat plates) attached to the specified corners as shown on Drawing No. 1000483. These bars are welded the full length of the tube.

The tubes with the bars attached are welded into cluster subassemblies per Drawing No. 1000484. Again, they are welded together the full length.

These clusters are then welded to other clusters and the base assembly as shown on Drawing No. 1000490, which is typical of the other rack size assemblies. These cluster attachment welds are again the full length of the tube.

ATTACHMENT 2

QUESTION NUMBER 2:

Provide a detailed description of the analysis or considerations used to establish that the tube inner compartment containing the Boral remains sealed against leakage. What are the potential consequences of pool water leaking into the area containing the Boral?

RESPONSE:

Consideration was given to maintaining the inner compartment containing the Boral sealed against leakage, and on the basis of the information available, it was decided to vent the Boral containing compartment and allow pool water to enter and exit without restriction.

The consequences of pool water in the area containing the Boral are discussed in the Brooks and Perkins' report, "The Suitability of Brooks & Perkins' Spent Fuel Storage Module for Use in PWR Storage Pool," Report No. 578 dated July 7, 1978, and did confirm the Boral panels are capable of meeting a forty year service life.

ATTACHMENT 2

QUESTION NUMBER 3:

What considerations have been taken to prevent off-gas from the Boral and swelling of the tube?

RESPONSE:

The off-gas from the Boral will not be a problem in a vented tube, thereby eliminating any swelling of the tube.

ATTACHMENT 2

QUESTION NUMBER 4:

Provide the basis for concluding that an empty rack will slide further than a full rack under seismic loadings. Provide a drawing of the equivalent stick model used in the sliding analysis and ind..ate the points where these loads were applied.

RESPONSE:

Since the spent fuel racks are stored under water, their seismic movements are caused by the horizontal inertia of "virtual mass" which is the sum of the body mass and the "hydrodynamic mass". The magnitude of the hydrodynamic mass depends on the shape of the rack body and the density of water, and so is independent of whether the rack is loaded or empty. The friction force resisting the seismic movement is proportional to the buoyant weight of the rack and its contents, but because of larger horizontal "virtual mass" per unit weight, the ratio of inertia force to friction force is more for empty racks. For this reason, it was concluded that empty racks will slide further than a loaded rack under seismic loadings.

Figure 4.1 shows the equivalent stick model used in the sliding analysis. Time history of SSE seismic movement was applied at Node 8 which represents the pool floor.

ATTACHMENT 2

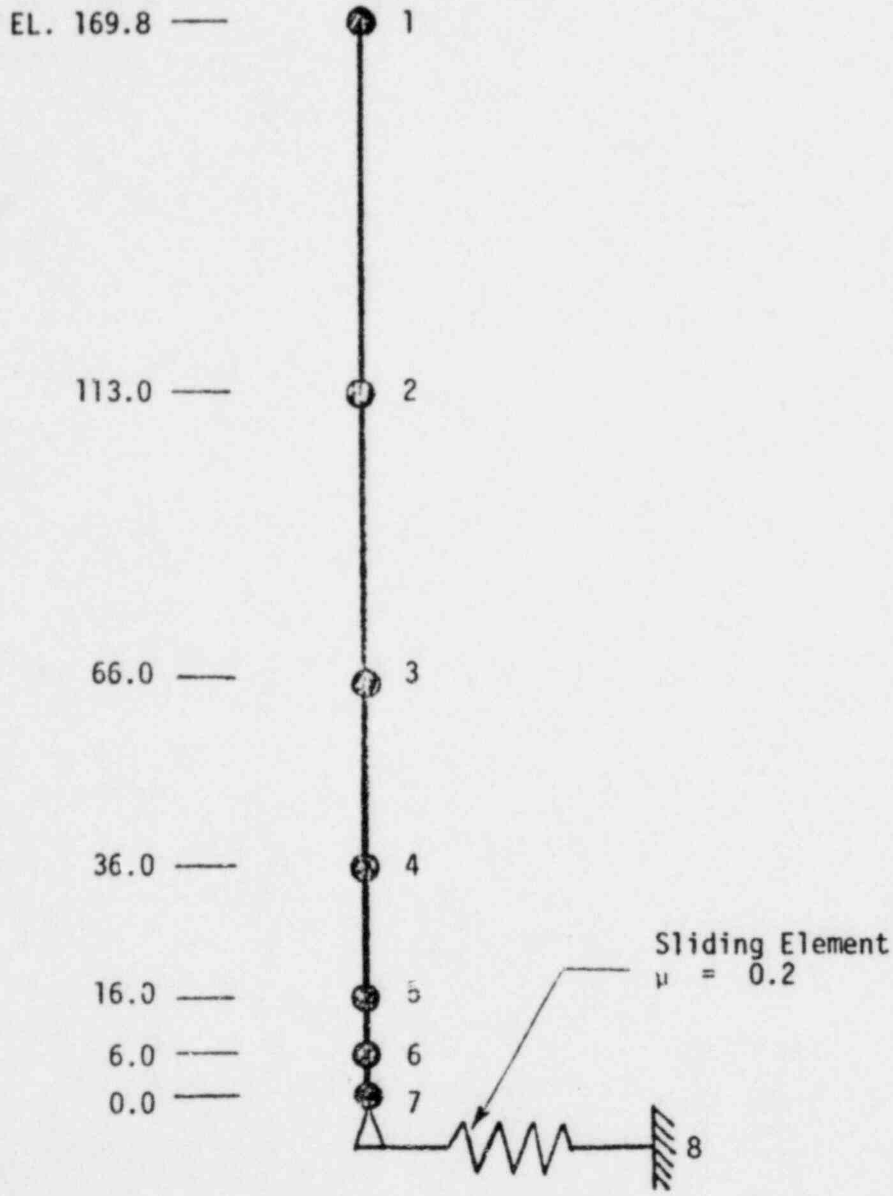


FIGURE 4.1 LUMPED MASS STICK MODEL FOR SLIDING ANALYSIS

ATTACHMENT 2

QUESTION NUMBER 5:

Provide the value of the "rattling factor" used in the seismic analysis.

RESPONSE:

Rattling factors account for the nonlinear effects of the fuel bundles moving within the spent fuel rack cells. The magnitude of these factors depend on the structural and damping properties of the rack and fuel bundles as well as on the level of excitation. The rattling factors used for the Zion rack evaluation ranged from 1.10 (for SSE loading of 10 x 11 size rack in the direction parallel to the longer side) to 2.57 (for OBE loading of 5 x 10 size rack in the direction parallel to the shorter side). These are upper bound factors computed using a conservative assumption that all the fuel bundles inside a rack "rattle" in phase.

ATTACHMENT 2

QUES...ON NUMBER 6:

Provide a description of the thermal gradient analysis considerations, include the thermal gradient considered and a discussion on why this was considered a conservative estimate of the worst case, i.e., the gradient between a full and empty cell.

RESPONSE:

The thermal gradient due to the placement of a hot fuel bundle in an empty rack is as shown in Figure 6.1 (0°F at the rack bottom and 32.38°F at the top). Stresses caused by this thermal gradient were computed using a finite element model which is also shown in Figure 6.1. To minimize the computation cost, only the central part of the rack body was modeled with the sides restrained from lateral translation, thus representing the worst case and predicting conservative stresses. It is important to note here that, near the top of the rack the thermal gradient, and hence, the resulting thermal stresses are maximum, but the dead load and seismic stresses are minimum. Thermal gradient near the bottom is very small where the seismic and dead load stresses are maximum.

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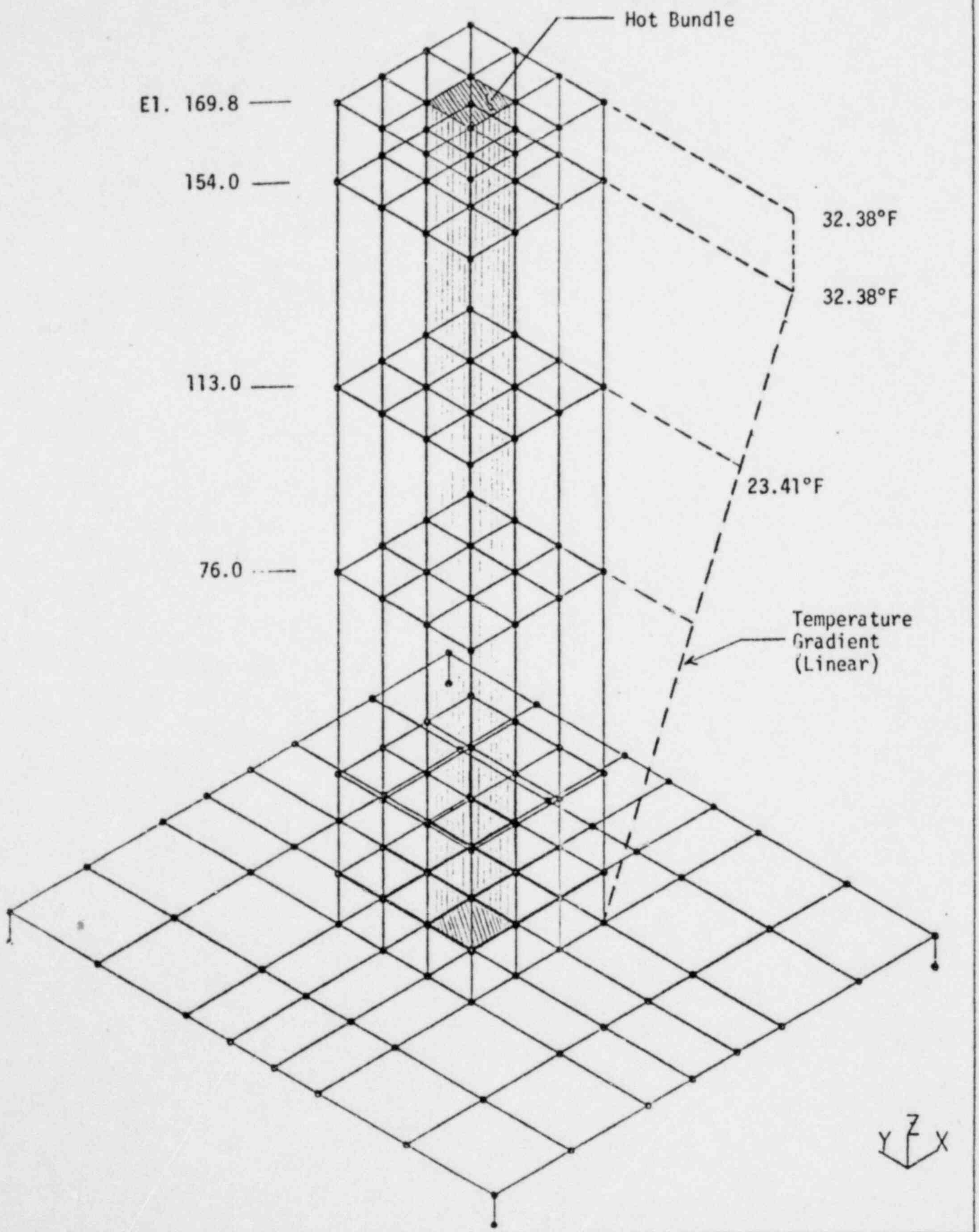

FIGURE 6.1 MATHEMATICAL MODEL FOR THERMAL (Q) ANALYSIS

ATTACHMENT 2

QUESTION NUMBER 7:

The fuel bundle drop analysis considered a drop at the most "critical" location on the rack, provide a description of this location and drawings to illustrate the postulated configuration of the fuel bundle at impact. Discuss the procedure for limiting the height of the fuel bundles above the racks to 24 inches. Discuss the consequences of a fuel bundle dropping straight through the tube and impacting the bottom of the rack.

RESPONSE:

The top corners of the racks were found to be the most critical locations for evaluating the consequence of dropping a fuel bundle. When the fuel bundle drops on the rack, the cross-sectional area of the cell walls absorbing the impact energy increases as the load is transmitted downward. Since this gradually-increasing cross-sectional area is minimum when the fuel bundle drops on a corner, the latter constituted the most critical location.

For evaluating the consequences of fuel bundle drop, the bundle configuration was assumed to be vertical at impact

(Figure 7.1). An inclined drop was judged to be less critical from the following considerations:

(a) The impact area will be larger,

(b) The impact will be "softer" because of the flexibility of the fuel bundle itself.

The length of the fuel handling tools and interlocks on the fuel pool bridge hoist limits the distance between the top of the rack and fuel assembly to less than 24." Fuel assemblies thus cannot be raised above the 24" limit. Consequences of the fuel bundle dropping straight through the tube and impacting the bottom of the rack have been invesitgated. The method of analysis and the results obtained are briefly described below:

The fuel bundle will drop approximately 164 inches from the top of the rack to the rack base plate. If the fluid drag on the bundle is neglected (a conservative assumption), the impact energy will be approximately 254,000 in-lbs. This energy will be absorbed by the following mechanisms:

(a) Since the fuel bundle is "soft" as compared to the rack, a large part of energy will be absorbed by the collapsing of the fuel bundle, thus limiting the maximum load transmitted to the rack.

ATTACHMENT 2

(b) A part of the enrgy will be absorbed in bending the base plate inside the fuel cell.

If it is conservatively assumed that, in the extreme case, the bending of the base plate causes a localized plastic hinge to form at the intersection of the tube wall and the base plate, the upper bound stress due to the accidental fuel bundle drop can be evaluated by applying at the cell wall the load required to form such a localized plastic hinge. This was done using a finite element model of a portion of the rack in the vicinity of the bundle drop. Loads were computed and stresses were determined at the bottom of the tube wall.

The poison material is capsulated at a height of 4.26 inches from the base plate. Maximum stress in the outer tube wall at that level was computed to be 18.9 ksi, well below the yield stress limit of the material. Also, it has been observed that the loads dissipated rapidly in the structural panels, indicating that the overall structural integrity of the rack is not impaired.

ATTACHMENT 2

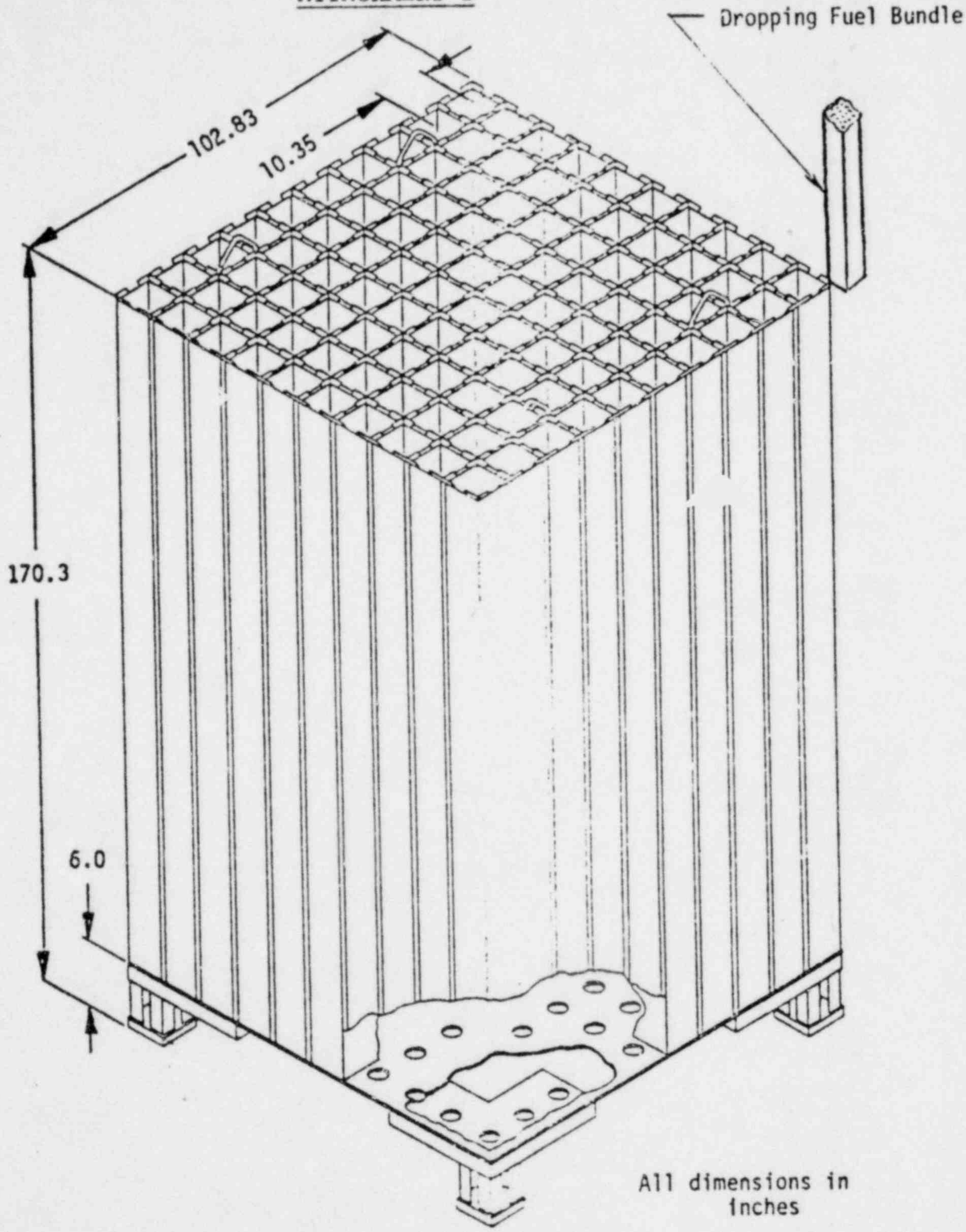

FIGURE 7.1 SPENT FUEL RACK SHOWING CRITICAL LOCATION OF POSTULATED FUEL BUNDLE DROP

ATTACHMENT 2

QUESTION NUMBER 8:

The results from the sliding analysis indicated that one rack could potentially slide 1.31 inches and that the minimum gap between any two adjacent racks is 2.4 inches. Discuss the basis for concluding that two adjacent racks could never slide out of phase, actually slide towards each other, and impact. They potentially could close a gap of 2.62 inches (180 degrees out of phase).

RESPONSE:

Each rack can potentially slide a distance of 1.31 inches towards each other. However, providing a space between the two adjacent racks less than twice this distance was justified from the following considerations:

(a) 1.31 inches is the peak movement of the rack obtained from a time history analysis. Under identical conditions, the adjacent rack would be in phase and would also move 1.31 inches in the same direction, in which case the original gap between the two racks would remain unaltered. However, since the adjacent rack is not likely to have identical conditions, the gap status is likely to change. Only if the two racks have identical conditions and their movements are exactly 180° out of phase, the minimum required gap to preclude impact would be the absolute sum of the movements of two racks, i.e., 2.62 inches. However, the probability of satisfying both these conditions simultaneously is extremely small, which justifies the use of a lesser gap. If SRSS method is applied to account for the low probability of the phenomenon, the required gap would be 1.414 times 1.31, i.e., 1.85 inches which is less than the 2.4 inches gap provided.

ATTACHMENT 2

(b) 1.31 inch is the predicted movement of the empty rack computed using the minimum coefficient of friction. It is judged that the loaded racks would slide significantly less than the empty racks. The reasons have been outlined in the response to Question No. 4.

ATTACHMENT 2

QUESTION NUMBER 9:

Provide the type or grade of stainless steel used in the construction of these racks.

RESPONSE:

All structural materials, with the exception of Boral, are stainless steel grade 304.