50-266

Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201

October 10, 1978

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Chief
Operating Reactors Branch #1

Gentlemen:

DOCKET NOS. 50-266 AND 50-301
ADDITIONAL INFORMATION
SPENT FUEL STORAGE EXPANSION
POINT BEACH NUCLEAR PLANT UNITS 1 AND 2

Mr. Schwencer's letters dated September 11, 1978 and October 3, 1978, forwarded a number of requests for additional information regarding our proposal to expand the spent fuel storage capacity at our Point Beach Nuclear Plant, Units 1 and 2. Our request for permission to increase the spent fuel storage capacity at Point Beach was submitted on March 21, 1978, and has been subsequently amended on June 14, 1978 and September 29, 1978. Your previous requests for additional information, dated June 5 and June 9, 1978, were responded to with our letter of July 19, 1978. Some of the questions in this current information request have asked for clarification of items discussed in our July 19, 1978 responses.

The attached responses have been numbered to match the respective information requests from your September 11, 1978 letter. These responses are prefixed by the letter C. Response number A-2 from our July 19, 1978 letter has been revised to address the concern identified in your October 3, 1978 letter. In addition, we have enclosed revisions to our previous responses numbered A-3, A-6, and A-28. These responses were revised to be consistent with the September 29, 1978 amendment to our license amendment request.

As specified, we are sending you three signed originals and, under separate cover, forty copies of these responses. Please contact us if you have any further questions concerning these matters.

Very truly yours,

Executive Vice President

Sol Burstein

7810230272 P

Enclosures

Question C-1

In regard to your July 19, 1978, response to question A-1, the fuel loading limit should be stated in terms of the grams of uranium-235; or equivalent, per axial-centimeter of fuel assembly. This is a more inclusive limit than one on the U-235 enrichment. A 44.8 gram per centimeter limit is acceptable for both the present racks and the modified racks even though the calculations for the present racks were done for a lower fuel loading. (i.e. 3.5% uranium-235).

RESPONSE

Licensees July 19, 1978, response to question A-1 was based on the respective criticality analyses covering storage of unirradiated fuel assemblies with a nominal 95% theoretical density. An enrichment of 4.0 w/o, (44.764 gm U-235 per axial cm.) was used in the analyses for the proposed fuel racks containing Boraflex poison material. For the existing storage racks which have a 15.5 in. center to center spacing, an enrichment of 3.5 w/o was used for earlier criticality analysis. Because the licensee has not performed the criticality analyses for the 15.5 in. center to center rack design using 4.0 w/o enrichment fuel (44.8 gms U-235 per axial cm.), the original limit of 3.5 w/o was retained in the proposed Technical Specification for the existing spent fuel racks.

Question C-2

In your response to question A-3, you state that Boraflex samples will be irradiated for ten years by the most recently discharged fuel assemblies. Describe how it is known that the radiation dose in any Boraflex plate in the pool could not be greater than that over the full life of these racks.

RESPONSE

The response to Question A-3 briefly outlined the poison material surveillance program.

The poison material gamma radiation exposure has been conservatively calculated for two different cases. Because of the number of storage positions (1502) provided by the new storage rack design and the nominal number of assemblies discharged into the pool each year (40/reactor/per year), approximately 17 years would be required to fill the pool (still maintaining core unload capability). As the present license is for 40 years if the pool were to be filled and the contents then shipped off-site, it is possible that all storage positions would see freshly discharged fuel assemblies on three occasions during a nominal 40 year plant operation. Thus, the base case was specified as 3 cycles of a freshly discharged assembly sitting in one position for 13 years (while 3 cycles is only 39 years, Unit 1 has already been operating for about eight years). The upper bound of the gamma radiation exposure was also calculated assuming a freshly discharged assembly was inserted in one position every six months (80 cycles). The calculated gamma radiation exposure of the poison material for these two cases is:

1.10×10^{12} rads for 80 - 6 month cycles

0.87×10^{11} rads for 3 - 13 year cycles

For the six month cycle case, the radiation exposure rate is 0.134×10^{11} rads per cycle. To attain the expected 0.87×10^{11} rads exposure at the six month rate requires 7.5 cycles or about 3.75 years. Thus, the sample extracted after the fifth year, which has seen 10 six month cycles, should adequately represent the lifetime exposure of three thirteen year cycles.

Question C-2 (cont.)

The tenth year sample point is included as a check.

Question C-3

In Section 2.2 (page 2-3) of revision 1 of the submittal, the following sentence appears, "The Neutron poison assembly consists of a 0.031 thick x 1 1/2" x 8 3/4" stainless steel inner can to which is bonded two sheets (0.100" x 8 1/2" x 145 1/2") of borated silicone rubber (See Appendix A)". Explain the following:

a. Why the thickness of the stainless steel next to the Boraflex in the PDQ calculational model, shown in Figure 3.1-1, is 0.035 inches?

b. Why the 1 1/2" dimension in the above statement is given as 1.64 inches in Figure 2.2-2A and a cumulative 1.822 inches in Figure 3.1-1?

c. Why the 0.031 inch thick stainless steel can appears to be 0.097 inches thick in Figure 3.1-1?

d. Why the 8 3/4" dimension in the above statement appears to be 8.288 inches in Figure 2.2-2A (page 2-6) and an accumulated 8.287 inches in Figure 3.1-1?

e. Why instead of two sheets of 8 1/2" wide Boraflex bonded to one 8 3/4" wide stainless steel can, as stated above, there is only a single sheet of Boraflex shown in Figure 3.1-1?

f. How can it be assumed that the Boraflex will remain bonded to the stainless steel when experimental data in Brand Industries Report Number 1047-1, which is referenced in Appendix A, show that under long term irradiation the Boraflex will shrink by more than two percent? Thus a 145 1/2" long piece of Boraflex will tend to shrink about 2.9 inches while the stainless steel will not.

RESPONSE

a) The .031 material was increased by a maximum thickness tolerance of .004" thus making the maximum material thickness .035", for calculational purposes.

b) The 1 1/2" dimension in the text was a nominal dimension that was not intended to be the actual manufacturing dimension. The 1.64" dimension in Figure 2.2-2A was only for the special poison boxes that are added in the inspection area; the 1.822 inches was due to discrepancies that arose during the time of design changes.

c) Here also, the design iterations at this time caused the above discrepancy. The calculational model originally contained an insert wall of .097 inches, whereas the text describes the revised design which identifies the poison cladding as .031 material.

RESPONSE (cont.)

d) The 8 3/4" dimension was a nominal dimension in the text and not intended as the actual manufacturing dimension. Also it was originally thought that the poison compartment would be dimensionally greater than the fuel assembly opening. The fuel assembly opening, 8.288 inches in Figure 2.2-2A was the mean dimension whereas the 8.387 inches in Figure 3.1-1 was intended as a maximum calculational dimension.

e) Around any fuel storage space, except at the rack periphery, there are four sheets of poison, even though two are actually located in adjacent storage positions. Each poison insert assembly contains two sheets of poison, one for the storage box it is in and one for the adjacent storage box.

f) In the initial design, the bonding was used only for assembly purposes. After the box was closed the Boraflex was contained by the sheets and bottom closure. Bonding has since been eliminated because it is not needed.

These questions have been resolved and clarified by the Applicant's recent submittal of Revision 2 to the application dated September 29, 1978. Minor dimensional changes have been made, including a reduction in the thickness of stainless steel can which encapsulates the poison material from 0.031 inches to 0.020 inches. The effect of manufacturing tolerances from a reactivity perspective is discussed in Section 3.3 of Appendix B to the application.

Question C-4

In Table 3.2-1, the overall dimension of the fuel assembly is given as 7.763" while the fuel region dimension in the calculational model on Figure 3.1-1 is 2 x 3.892 or 7.784". What is the basis for the 3.892 inch dimension in the calculational model?

RESPONSE

The 7.784" dimension was arrived at by multiplying the pitch dimension (in the same table) of 0.556 times 14. This number rather than the 7.763 was originally used in the calculations.

Question C-5

What is the chemical composition of all of the 4.742×10^{21} atoms of Boron Ten per cubic centimeter of Boraflex, which is stated on Figure 3.3-2 to be the minimum Boron Ten density?

RESPONSE

The stable isotope Boron Ten which occurs naturally as a small percentage of the element Boron, is in Boraflex as Boron Carbide, B_4C. The design is based on a minimum B_4C loading of 37.1 weight percent in the Boraflex. The weight percent Boron in the B_4C powder is assumed to be 70%. Chemically, Boron Ten is indistinguishable from the element Boron.

Question C-6

Provide experimental data that show that the relatively high dose rates, which were used for the irradiation tests, have as much of an effect on the Boraflex plates as the lower dose rates that will be encountered in the spent fuel pool.

RESPONSE

The exposure rate in the pool varies from about 2.5×10^5 rads/hour (realistic) to 3×10^6 rads/hour (maximum): these are simply the total exposure for the two cases analyzed (see the response to Question C-2) divided by time.

The test data contained in BISCO Report 1047-1 was obtained at the University of Michigan. The test rates were about 10^5 rads/hour for the Cobalt 60 testing (report Attachment 3) and about 3×10^7 rads/hour for the reactor testing (report Attachment 4). These rates are actually above and below the expected exposure rate in the spent fuel pool but not by a significant amount.

Question C-7

Indicate the various materials used in the rack construction. If any 17-4 PH stainless steel components are being used, provide the heat treatment temperature(s), and indicate that the components will be hardness tested and either pickled or grit blasted in order to remove the surface film resulting from the heat treatment.

RESPONSE

All materials used in the rack construction are type 304 stainless steel and the Boraflex poison material. 17-4 PH stainless steel is not used.

Question C-8

On page 5-1 of your submittal load combination (4) should have a multiplying factor of 1.4 applied to D. Also, the following additional load combination must be satisfied: $D + L + T + 1.25E_o < U$.

RESPONSE

The omission of the multiplying factor 1.4 applied to D in the load combination (4) on page 5-1 of our submittal is a misprint. The correct load combination $U=0.75(1.4D+1.7L+1.7T)$ has been considered in the design.

A check has been made to ensure that the additional load combination $U>D+L+T+1.25E_o$ is satisfied and could be included in the list of equations considered.

Question C-9

Provide details of the fuel bundle impact analysis considering the impacts which occur during a seismic event. Include the generated loads, the gross and the local stresses on the racks, a description of one model used in the anlaysis, and the assumed velocity of the fuel assembly. Indicate the loading combination utilized to consider this effect and the corresponding acceptance criteria. Also, demonstrate that the fuel assemblies themselves will retain their structural integrity and will not suffer cladding damage as a result of impacting the can.

RESPONSE

Each fuel element is simply supported at its bottom. During seismic excitation the rack will move and impact the fuel. Calculations have shown that an empty fuel rack submerged in water will have very little motion relative to the pool, i.e., it will move with the motion of the pool. There is hydrodynamic coupling between the fuel element and the rack, and as this water gap thickness goes to zero this coupling force will become large, the effect being nonlinear. This is not accounted for in the analysis and, as a result, the impact forces are considered to be overestimated.

It is assumed that one fourth of the weight of the spent fuel elements impacts the top of the rack assembly. The velocity of impact is considered to be the velocity equivalent to the spectrum value for the lowest natural frequency of the rack, excluding the mass of the spent fuel, but including the entrapped water, the poison packages, and the hydrodynamic weight. The effect of the hydrodynamic coupling between the rack and the pool wall is included.

The kinetic energy of one fourth of the weight of the spent fuel elements having this velocity is considered to be absorbed as strain energy in the rack assembly. This is converted to an equivalent force, treating the rack assembly as a beam, and including shear deformation.

Because of the low spectrum value at the lowest natural frequency, which is 23 hz in the East-West direction, the relative velocity is only 0.17 in/sec. and the corresponding force due to impact is 4,100 lb.

RESPONSE (cont.)

The shear force in the empty module is 1,000 lb. so that the total load from this model is 5,100 lb. These values are for the OBE; the values for the SSE would be double this. The lowest natural frequency in the North-South direction is 30 hz. and the forces are smaller than for the East-West direction.

These impact loadings are approximately half of those generated by seismic forces in the rack and are thus not used for rack stresses. The acceptance criteria is the stress in the storage box on the internal structure. In accordance with the NRC Standard Review Plan and allowable stress as delineated in the Response to Question C-12, the maximum stress is 2,240 psi and the strength ratio is 8.9.

Question C-10

Indicate how the additional mass of water was accounted for in the seismic analyses. Provide details of the method used to calculate the virtual mass.

RESPONSE

The virtual or hydrodynamic mass was calculated by a method similar to that described in the literature*. The mathematical model consisted of a rectangular parallelepiped which represented the rack inside a rectangular cavity which represented the pool. Equations were set up to describe the motion of the fluid as the parallelepiped moved inside the cavity. Because the gap between the two is much smaller than the dimensions of the rack it was assumed that the velocity is uniform across the gap. It was also assumed that the fluid motion is horizontal, the effect of fluid moving in the vertical direction being neglected. In addition, elastic deflections of the rack itself were neglected. As a result, the calculated hydrodynamic mass is higher than the actual value. The calculations followed the method described in the reference.

The calculated value of the hydrodynamic mass was verified by an experiment** in which a structural rectangular tube was used to model the parallelepiped. A plexiglas pool surrounded the tube with a gap of 0.5 inch on all sides. The tube was mounted on two steel cantilever springs which, because of the parallelegram effect, constrained the tube to move horizontally without rotation. Seals at the top and bottom of the tube restricted the vertical fluid flow. The system was vibrated horizontally by an electromagnetic shaker, tests being made both with and without fluid over a wide frequency range. Calculations showed that measured amplitudes, with fluid present, were consistent with a fluid coupling between the tube and the pool wall as calculated by the theory. This is considered to be good verification of the calculations.

Question C-10 (cont.)

*R. J. Fritz, "The Effect of Liquid in the Dynamic Motions of Immersed Solids", Transactions ASME, journal of Engineering for Industry, February, 1972.

**E. F. Radke, "Experimental Study of Immersed Rectangular Solids in Rectangular Cavities", Project for Master of Science Degree, University of Akron, Akron, Ohio, 1978.

Question C-11

Provide details of the seismic analyses performed on the racks. Provide sketches of the math models utilized. Also, provide the frequencies up to 33 Hz, the corresponding mode shapes and participation factors. Indicate the damping values used and discuss the applicability of Regulatory Guides 1.60 and 1.61. Discuss the methods utilized to consider the effects of the gaps between the restraints and pool walls, and indicate whether the racks are physically connected to one another or if it is possible for adjacent racks to impact. Also, provide a summary of the resultant highest stresses and strength ratios, and the corresponding locations.

RESPONSE

The mathematical model is shown in Figure B.1. In this model masses M1 through M10 each represent one-tenth of the mass of the following: a) the rack structure, b) the fuel bundles, c) the poison assemblies, d) the entrapped water and e) the hydrodynamic mass. Flexible elements EL1 through EL11 represent the bending flexibility and the shear flexibility of the rack itself, considered as a beam. EL12 represents the local flexibility of the bottom part of the rack structure. The support between M1 and M2 represents the support that is placed between the rack and the pool wall to prevent horizontal motion of the rack relative to the pool wall. The fluid coupling element between each mass and the pool wall represents the effect of the motion of the pool wall on the rack structure due to hydrodynamic coupling. Calculations similar to those done in determining the hydrodynamic mass were performed and are similar to those described by Fritz, except that in the present case the pool wall is considered to move with the motion of the ground. It was found that the effect of this coupling can be accounted for by modifying the model participation factor in the dynamic design method.

The damping values used for the rack structural analyses were less than those suggested by Table 1 of Regulatory Guide 1.61 dated October 1973 which is a conservative approach. The seismic response spectrum curves used were those developed for the original design and analysis of the Point Beach Nuclear Plant. The curves are based upon a 0.06g OBE and 0.12g SSE ground

Question C-11 (cont.)

acceleration and account for the building flexibility; the vertical accelerations are considered to be 2/3 of the horizontal acceleration at all frequencies.

The calculations were done with the CTAC-MODE code. The CTAC portion of the code performs the flexibility analysis, generating a flexibility matrix for the model, while the MODE portion combines this matrix with the mass matrix (M) and finds the frequencies and mode shapes for the system. The normal mode method is then used to calculate effective forces on the masses, and these are applied to the structure to find the bending moment and shear force in each flexible element. The CTAC-MODE program was developed at the Bettis Atomic Power Laboratory. It has been used for many years and has been thoroughly verified.

Only one frequency each in both the N-S and E-W direction are below 33 Hz. The first three frequencies in the N-S direction are 22.48, 78.32, and 143.59 Hz. The first three in the E-W direction are 19.44, 66.09, and 119.72 Hz.

The mode shape ($\bar{x}_1$) is a 1 x 10 vector which has the maximum value equal to 1 and is as follows:

$$\bar{x}_1 = \begin{bmatrix} -.007 \\ .098 \\ .208 \\ .327 \\ .451 \\ .574 \\ .694 \\ .807 \\ .091 \\ 1.000 \end{bmatrix}$$

Question C-11 (cont.)

The participation factor is defined as:

$$P_1 = \frac{\overline{x}_1^T \, (M) \, \overline{C}}{\overline{x}_1^T \, (M) \, \overline{x}_1}$$

where superscript T denotes the transpose of the vector.

The vector $\overline{C}$ is a vector which describes the fluid effects of the system including fluid pool interaction effects based on mode shape with maximum value of 1. For this case:

$$\overline{C} = \begin{bmatrix} .508 \\ .508 \\ .508 \\ .508 \\ .508 \\ .508 \\ .508 \\ .508 \\ .508 \\ .508 \end{bmatrix}$$

Note: All 10 values are equal to .508

and P_1 = .705.

While the racks are not physically connected to each other, the racks are to be installed so that they are in contact with the adjacent racks. The highest stresses are rack internal and are reported along with strength ratios in the Response to Question C-12.

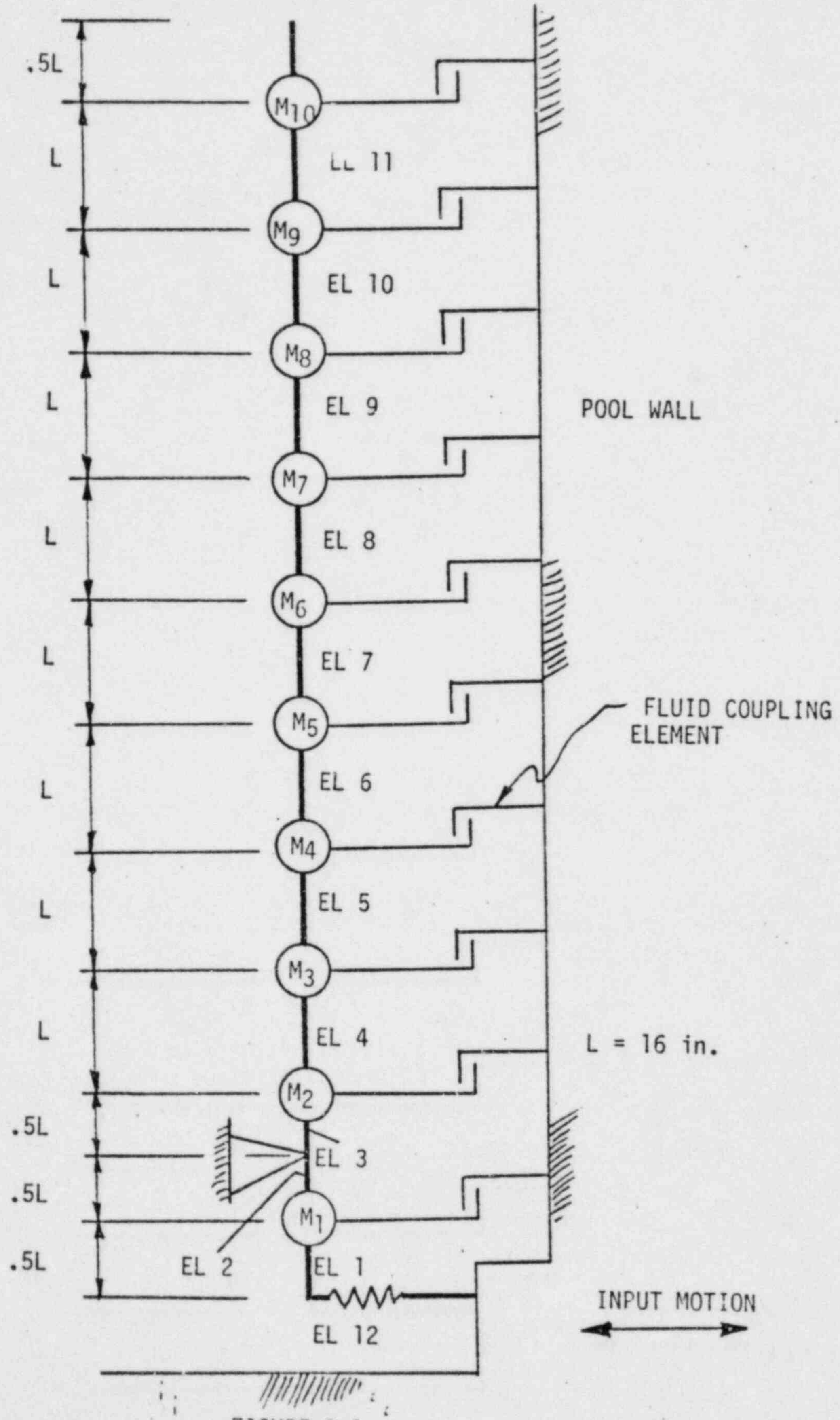


FIGURE B.1

Dynamic Model of the Spent Fuel Rack Assembly with Fluid Coupling to the Pool Wall.

Question C-12

It is stated that the structural analysis of the racks has considered all the loads and load combinations specified in the USNRC Standard Review Plan. Provide the loads and the load combinations and the corresponding acceptance criteria for the analysis of the racks. Also provide the highest stresses and strength ratios calculated for each combination. Verify that the material properties were derived at the appropriate temperature.

RESPONSE

The following is in accordance with the USNRC Standard Review Plan, Section 3.8.4, "Other Seismic Category Structures" dated November 24, 1975. Plastic design methods are not used for this equipment and of course only those evaluations identified for steel structures are applicable.

The loadings are defined as presented on pages 3.8.4-5 and -6 of the Review Plan. However, because this equipment is to be located in a spent fuel pool, many of the loadings are not applicable. This includes W, W_t, Pa, Ta, Ra, Yr, Yj and Ym. The following loadings apply to the largest rack (110 storage positions):

D = 39,974

L = 140,316

E = 10,817

E^1 = 21,634

Because of the nonapplicable loading parameters, the equations of pages 7 and 8 of the Review Plan can be reduced to the following:

Equation Number	Load Combinations		Acceptance Criteria
(1)	D + L	= 180,290	*S = 20,000 psi
(2)	D + L + E	= 191,107	S = 20,000 psi
(4)	D + L + To + Ro + E^1 (To & Ro = 0)	= 201,924	1.6/S = 32,000 psi

Question C-12 (cont.)

*Value for S is the allowable stress for Class 2 and 3 components for SA-240, 304 plate, at 200°F. from ASME-BPVC Section III - Division 1, Table 1-7.2, 1977 edition.

Internal rack stresses, arrived at by the SRSS method, and the strength ratios are as follows:

Equation Number	Maximum Stress	Strength Ratio
(1)	2,497 psi	8.01
(2)	3,610 psi	5.54
(4)	4,559 psi	7.02

confined to the upper 2" of the box, thus reinforcing the concept that the impact damage will be highly localized. The total energy absorbed by the box sections was 715-760 ft-lbs. of energy. While the experiment was simple and the loading slowly applied, the results are considered to be applicable to a worst case rack deformation evaluation. With an impact loading situation it is expected that less deformation would occur because generally material strength, and ability to absorb energy, are greater at higher loading rates.

From the standpoint of maximum rack damage (deformation), a fuel assembly can be dropped on an outside, empty corner box wall; this represents the worst case whether or not the fuel assembly impacts straight (vertical) or inclined. The tests results suggest that 730 ft-lbs. of energy per .9" is absorbed. Employing this conservative value, a dropped fuel assembly would locally deform a storage box wall approximately 5 1/4 inches. This assembly was dropped from 3 feet, weighed 1,425 lbs., impacted with an energy of 4,275 ft-lbs., and achieved a maximum velocity of 14 feet per second (treated as a free-fall in air). Since an inclined drop would strike more than one box, it will generate less damage and less force per box than the straight drop case, and is thus considered as a less-severe case. The maximum load generated by the fuel assembly impact on the box wall will be approximately 20,000 lbs., a load which does not approach the ultimate strength of the box-to-box attachments welds of about 360,000 lbs. Since the loads are less than the combination of normal plus SSE loads, the effects on the fuel pool liner and floor are less than the design conditions.

Since a fuel drop accident is not restricted to occur only during seismic or loss-of-cooling events, the loads of impact are combined with the normal loads in order to evaluate the effects on the fuel racks. The damage

Question C-13

It is our position that the effects of a dropped fuel assembly must be considered. Discuss and quantify the local and gross affects on the racks, fuel pool liner and floor for the following three cases of a dropped fuel assembly:

a) a straight drop on the top of a rack

b) an inclined drop on the top of a rack

c) a straight drop through a can with the fuel assembly impacting the bottom of a can at both a flexible location of the rack and right at a vertical support location.

Include the kinetic energies, drop height, maximum velocities, mass, and assumed ductility ratios. Indicate which load combination these loads are included in. Also, indicate whether a fuel assembly will develop the highest kinetic energy of all objects that are handled and could possibly be dropped into spent fuel pools.

RESPONSE

A dropped spent fuel accident was considered during the analysis and design of the new spent fuel racks.

Due to the nature of compressive elastic and plastic box buckling and the inability of plastic strain theories to accurately and adequately deal with compressive plastic buckling, two full-size box sections have been experimentally tested to ascertain force and deformation characteristics. A single, unsupported box section 16" long was inserted between the head and anvil of a 120 kip compression testing machine and deformed (see Figures 1 and 2). The maximum resisting force of the two box samples tested was 16 to 20 kips, achieved at a deformation of 100 mils. The box corner then began to crush in an accordion-like manner (see Figures 3 and 4). Due to the loading on the corner of the box, the seam weld which joins the box walls began to rip. The crush test was continued to a deformation of greater than 900 mils. The force varied generally as shown on Figure 3. A significant outcome of the test was the absence of any fastfailure mode: the weld seam tore locally and no cracks were propagated in advance of the weld tear. Given that the maximum deformation of the test went to 0.911 the second significant outcome was that all the buckling damage was

Question C-13 (cont.)

to the lead-in guide and box wall done by this type of accident is highly localized, except at only one location each in the two pools. In the North pool only the N-W corner is without a lead-in guide, which is the type of corner used in the above test. In the South pool, only the S-E corner is like this. The lead-in guide deformation, which was not included in the test, would absorb some of the impact energy which would reduce the amount of box wall deformation as stated above. There is a low probability of these two corners being the target for such a drop. The remainder of the rack is unaffected in either form or function.

In addition, if a fuel assembly were to be dropped and impact upon an intersection of storage positions, the deformation would be much less because the surrounding storage positions would also absorb energy.

The probability of a straight drop of a fuel assembly directly into an empty fuel position is much more remote than any other drop accident; it is most likely that the fuel assembly would strike a lead-in guide and be deflected so that energy would be dissipated in deformation of the lead-in guide and within the fuel assembly itself.

For the drop of a fuel assembly directly into a storage position, because the clearance between fuel dimensions and box dimensions are quite close the fuel assembly would become a leaky piston and the fuel box would become a leaky cylinder. The hydraulic forces generated when the fuel assembly initially enters the fuel box are quite large and would serve to retard the fuel assembly during the next 14 feet of its descent. The velocity of impact has been estimated from hydraulic considerations of the leaky piston and cylinder to be in the range of 20 to 25 feet per second (the equivalent "free-fall" velocity would be 33.1 feet per second). The energy of impact is from 8,860 to 13,900 ft-lbs (as compared to a "free-fall" energy of 24,300 ft-lbs). Consideration of the

Question C-13 (cont.)

hydraulic effects reduces the energy of impact to 40% - 60% of the "free-fall" value. When the fuel assembly impacts the bottom plate, it is postulated that all 196 fuel and water rods buckle at the Euler buckling load. This load is calculated to be approximately 260 lbs. per rod, or 51,000 lbs. per assembly. The ultimate strength of the welds which secure the bottom plate onto the fuel box is about 5,600 lbs. per linear inch (by actual testing) and the storage position bottom plate is surrounded by about 39 linear inches of weld. Consequently, the ultimate strength is about 218,000 lbs. The conclusion is that the bottom plate remains in the box and no damage would occur to the liner plate or the floor.

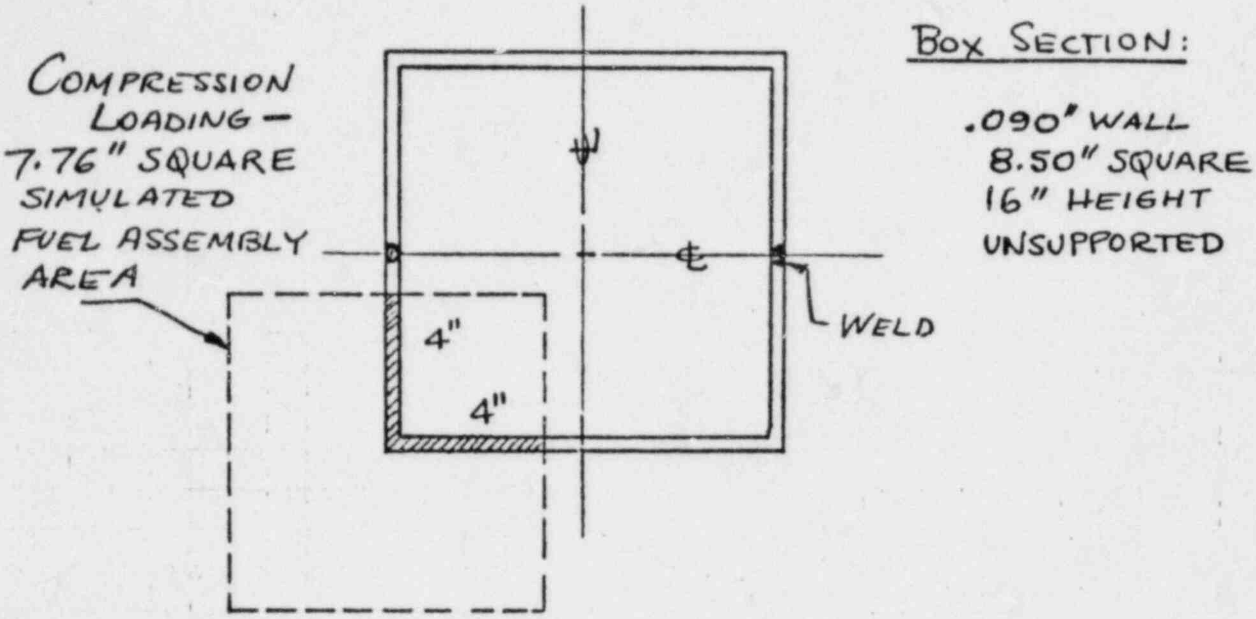


FIGURE 1

CRUSH TEST LOADING – TOP VIEW

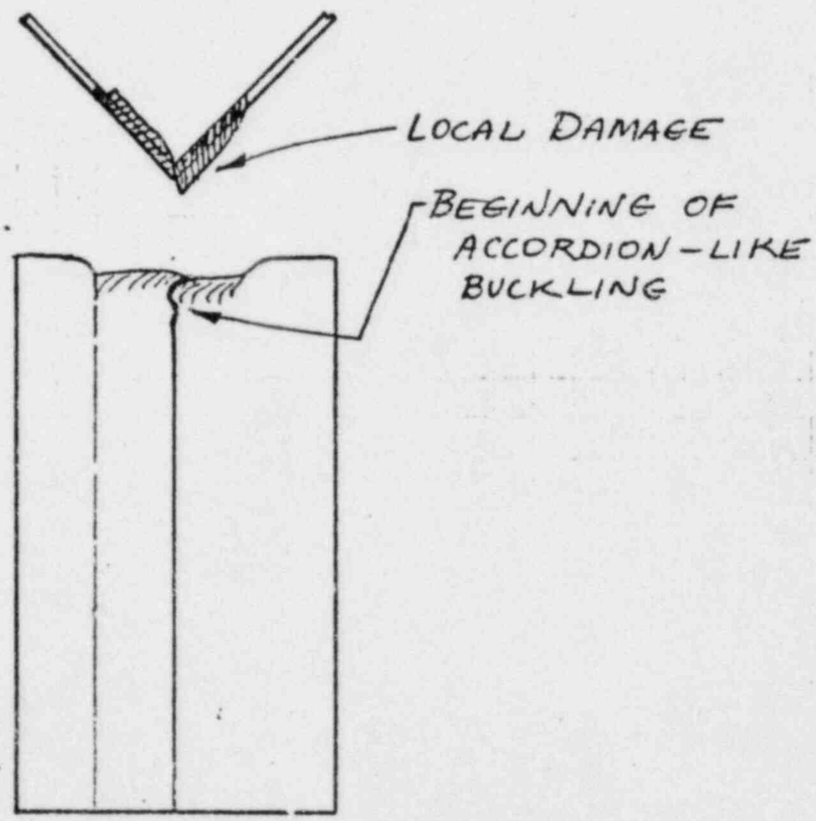


FIGURE 2

SKETCH OF LOCALIZED DAMAGE

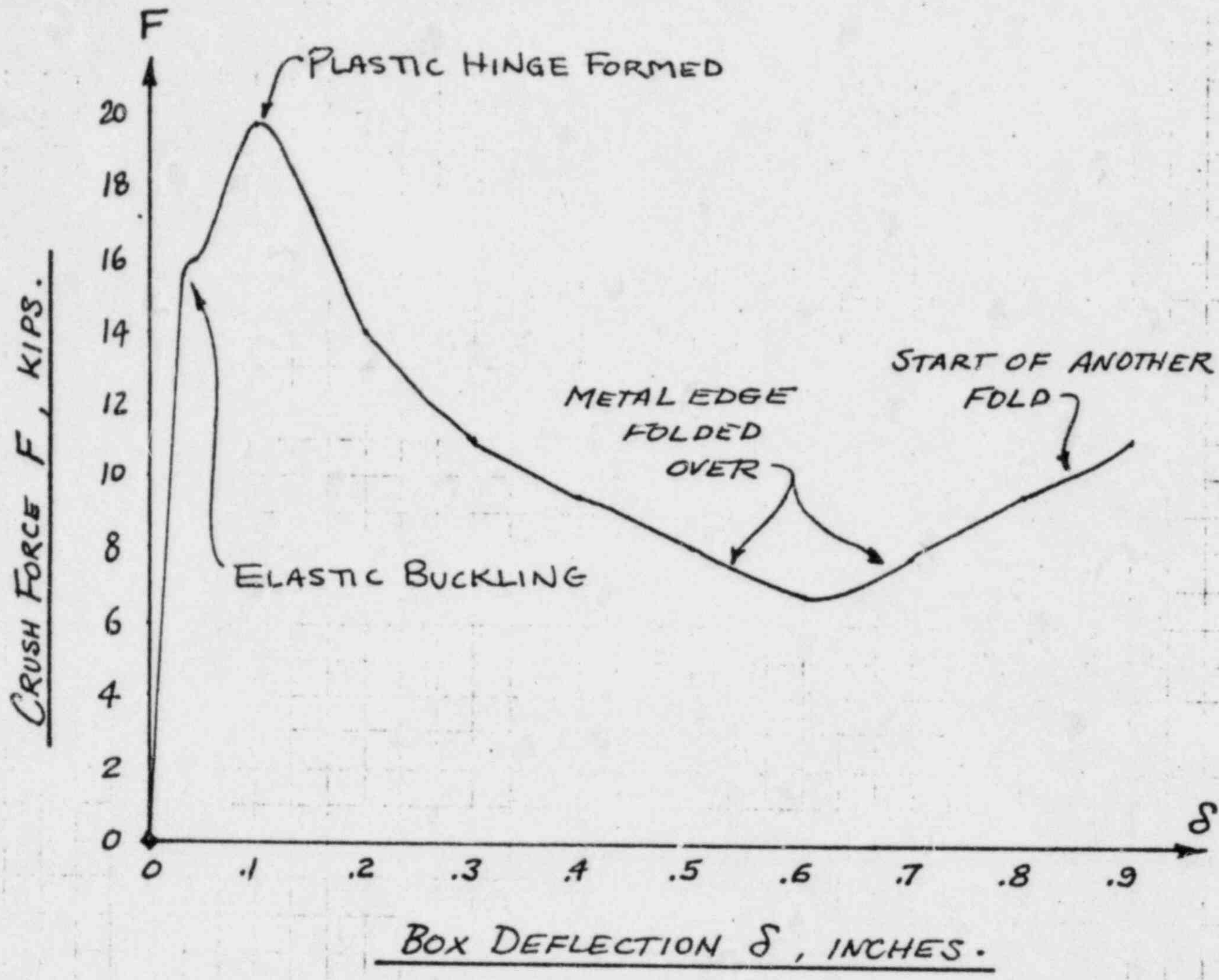


FIGURE 3

BOX CRUSH TEST RESULTS

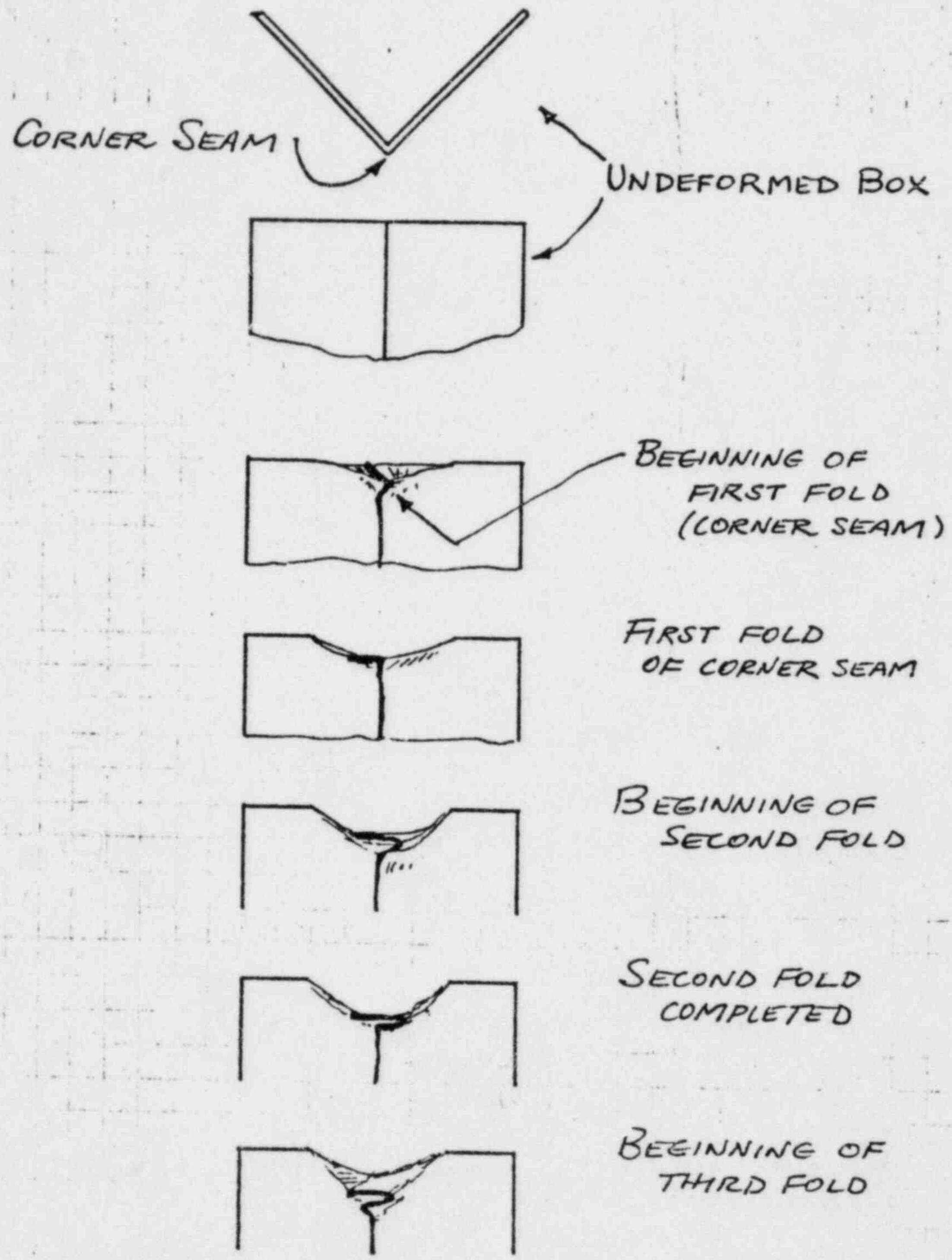


FIGURE 4

PLASTIC BUCKLING AND DEFORMATION BEHAVIOR OF CORNER SEAM

Question C-14

The effects of a stuck fuel assembly should also be considered. The resultant loading on the can and overall rack and support should be quantified. Indicate the maximum uplift force available from the crane. Also, indicate which loading combinations include these effects.

RESPONSE

The maximum uplift force that can be exerted on a fuel assembly is 2,000 pounds. A load cell in the spent fuel pool bridge hoist is used to deenergize the motor when this load is exceeded.

The upward loading only serves to reduce the downward loading on the rack, and is not sufficient to lift any storage rack in the pool even if the rack does not contain other spent fuel assemblies. The loading has been considered in the rack design but is insignificant.

Question C-15

Discuss the ability of the Boraflex plates to maintain their structural integrity during vibratory environments. Indicate whether they can come unbonded during an earthquake and be subjected to increased loading. Are the plates counted on to carry any loading in their normal configuration. Discuss the possible loss of mechanical strength resulting from the environment, including irradiation, seen by the poison plates.

RESPONSE

Each Boraflex sheet is enclosed between two sheets of stainless steel. The Boraflex is a flexible, solid, stable material that is not free to escape or change position in a vibratory environment. Originally, bonding was specified only as an assembly procedure. This has since been eliminated. The Boraflex sheets are not counted on to carry any loading. The stainless steel sheet cladding and lead-in guide assembly carries all loading. Boraflex I retains its mechanical strength in the environment, including irradiation, seen by the poison plates and is so stated in the Response to Question A-12 which refers to supporting tests conducted by BISCO, the manufacturer.

Question C-16

Provide details of the pool modification phases and indicate whether all racks will be seismically supported during all phases.

RESPONSE

The Point Beach Nuclear Plant Spent Fuel Storage Expansion modification consists of two (2) phases; modification of the North pool (planned for the Summer of 1979) followed by modification of the South pool (planned for the Summer of 1980). Because of the present storage situation at Point Beach, it will not be necessary to have any fuel assemblies in the pool being modified and thus all fuel assemblies will be stored in seismically supported racks during this project. This was also alluded to in our response to Question A-8.

The administrative procedures discussed in our response to Question A-8 will be followed as applicable to the sequence of events of this project. In addition, procedure PBNP 3.20, "Instructions For Processing Modification Requests", of the Point Beach Nuclear Plant Administrative Control, Policies and Procedures Manual will apply to this plant modification.

The general sequence of events for both phases of this modification are as follows:

1. Move all fuel assemblies and tools to the opposite pool.
2. Operate the pool cooling and purification systems to obtain and maintain an acceptable (for the diver) water temperature and maximize water cleanup.
3. First dive. Remove rack restraints and install lifting slings on all racks. The slings are anchored up out of the pool for the actual (later) rack removal. Check pool dimensions.
4. Remove racks one at a time. Clean and package for disposal.
5. Receive and inspect new pool equipment. Prepare the equipment for installation (additional cleaning, any required assembly, arrange

<u>Question C-16</u> (cont.)

for sequence of handling, etc.) <u>Second dive</u>. This will start the installation phase and will encompass all of the remaining events.

6. Survey the pool, remove any loose objects and generally prepare the pool.
7. Install the rack bases in the correct position and level.
8. Install the first rack and relevel as necessary.
9. Install the first rack seismic restraints.
10. Repeat steps "7", "8", and "9" for the remaining equipment.
    a. check all design "gaps".

Question C-17

Provide the water chemistry which will be maintained in the spent fuel pool. Include the boron concentration, ph, chloride, fluoride and any heavy metal concentrations.

RESPONSE

The fluoride and chloride limits in the spent fuel pool water are 0.15ppm maximum. This limit is based upon an identical limit in Technical Specifications for the Reactor Coolant System because during refueling the spent fuel pool coolant mixes with the primary coolant. By imposing the 0.15ppm limit, there will be no unacceptable contamination of the reactor coolant system.

The boron concentration has a minimum limit of 1800ppm by the plant Technical Specifications.

There is no limit on heavy metals and the water is not analyzed for them.

The spent fuel water is required to be sampled monthly for boron concentration by the plant Technical Specifications. The practice at Point Beach Nuclear Plant is to sample the pool water approximately every week.

The ph may range from 4.5 to 5.1 and is controlled primarily by the boron concentration.

Question C-18

Provide a summary of the original design criteria, i.e., PSAR, used in evaluating the piles. If this differs from criteria now being used i.e., AISC and 0.9 x pile yield strength, provide technical justification for using this criteria. Also, provide a summary of the pile strength ratios corresponding to the most critical loading combinations.

RESPONSE

The following is a summary of the original design criteria per the FFDSAR and the design criteria being used for the present evaluation. Any variances are noted and the technical justifications are provided.

Question C-18 (cont.)

No.	Item		Original Design Criteria per FFDSAR	Design Criteria for Present Evaluation	Variance	Remarks
1.	Design Method		Working Stress Design Method	Working Stress Design Method	No	
2.	Yield Stress		55 ksi	63.9 ksi	Yes	See Technical Justification (1)
3.	Combined Stresses and Allowables	OBE	Due to dead, live and OBE loads			
			max. axial compressive stress fa < 12000 psi	max. axial compressive stress fa < 12000 psi	No	
			max. axial compr. stress + bending stress fa+fb < 33000 psi	max. axial compr. stress + bending stress: stresses combined per Sec. 1.6.1 and allowables (without one-third increase) per Sec. 1.5.1 of "Manual of Steel Construction," A.I.S.C. 7th Edition	Yes	See Technical Justification (2)
		SSE	$Y > \frac{1}{0.9}(1.0D+1.0T+1.0P+1.0Es)$	$Y > \frac{1}{0.9}(1.0D+1.0T+1.0P+1.0Es)$	No	

Question C-18 (cont.)

Technical Justification(1):

Mill Test reports for the installed piles indicate that the actual minimum yield stress of the ASTM A572-66, Grade 55 piles is 63.9 ksi. Therefore the actual minimum yield stress of 63.9 ksi is used in the present evaluation.

Technical Justification(2):

Axial compression + bending:

For the combined axial compression and bending stresses, the interaction formulas of Sec. 1.6.1 of "Manual of Steel Construction" AISC, 7th edition are used because they consider the fact that depending upon the conditions, the allowable stresses for axial compression and bending can be different.

The allowable axial compression stress is determined according to the slenderness ratio. The allowable stress for the strong-axis bending is conservatively taken as $0.60F_y$. Although the piles (14BP117) are non-compact sections, due to the lateral restraints provided to the compression flanges by the confinement from the surrounding soil, the allowable bending stress for the weak axis bending is determined according to Sec. 1.5.1.4.3 of "Manual of Steel Construction".

The following is a summary of the pile strength ratios for the most critical loading combinations:

No.	Item	*Strength Ratio	Remarks
1.	OBE: Axial Compression only	0.93	O.K.
2.	OBE: Axial Compression + weak axis bending	0.92	O.K.
3.	OBE: Axial compression + strong axis bending	0.67	O.K.
4.	SSE: Axial compression + weak axis bending	0.91	O.K.

* The strength ratios are based on the actual minimum yield stress of 63.9 ksi.

$$\text{Strength Ratio} = \frac{\text{actual stress}}{\text{allowable stress}} \quad \text{for OBE}$$

$$= \frac{\text{actual stress}}{(.9)\ \text{yield stress}} \quad \text{for SSE}$$

Question C-19

Provide the water chemistry which was maintained in the pool used for storage of the Point Beach fuel assemblies which are to be placed back into the Point Beach Pools. Describe the examinations both non-destructive and destructive, which are planned to be performed on these assemblies to assure that there have been no deleterious effects which have resulted from this off-site storage.

RESPONSE

The water chemistry of the storage pools at both the Nuclear Fuel Services, Inc. West Valley site and the General Electric Company Morris site is characterized as demineralized water. This chemistry is maintained by running pool water through demineralizers utilizing ion exchange resins (which are periodically replaced). Water temperature is maintained between 25 and 40^{o}C at the Morris site and between 70 and 79^{o}F at West Valley.

All fuel assemblies will be visually inspected prior to loading into a cask for shipment. No destructive examination is contemplated.

There are no present plans or schedules to return spent fuel to Point Beach from existing off-site storage facilities.

Question C-20

Are examinations, destructive or non-destructive, planned to be performed on portions of the racks (i.e., weld sensitized areas, bolts or any other components of the racks which sustained similar high stresses during their residence in the Point Beach spent fuel pool) which you propose to remove. If so, please describe.

RESPONSE

No program of examination is contemplated for the racks which are being removed. Generally speaking, the loadings and accompanying stresses would be a maximum during a seismic occurrence but still within allowable safety margins for the material. As there have not been any seismic occurrences at the Point Beach Nuclear Plant during its lifetime, the stresses that actually existed in the racks were probably insignificant with respect to the material capability. Thus, it is highly unlikely that examinations would reveal anything. To perform examinations while not expecting to find anything would violate ALARA radiation exposure criteria.

Question A-2

NRC procedures require that an onsite test be performed to verify, within 95 percent confidence limits, that a sufficient number of neutron absorbing plates (i.e., poison plates) in the installed racks contain the required boron content to maintain the $k_{eff} \leq 0.95$. When the poison plates are made an integral part of the racks and the condition of the poison plates is continually monitored by surveillance tests, the NRC finds that a single initial neutron attenuation test on the racks is sufficient. However, a single initial neutron attenuation test will not be sufficient for the proposed racks with the removable poison inserts. Describe how you propose to periodically perform tests to verify, within 95 percent confidence limits, that there will always be a sufficient number of poison plates which contain the required boron content to maintain the $k_{eff} \leq 0.95$ in the proposed storage racks with the removable poison inserts.

RESPONSE

The rack design provides two semi-permanent poison assemblies per fuel assembly. These poison assemblies are locked in place by a lock bolt that cannot be removed without mechanically deforming the locking element. The lead-in funnel at the top of the fuel cell location is an integral part of the poison assembly and thus the presence of the neutron poison plates can be verified visually. The presence of the neutron poison plates will be verified visually prior to the installation of the storage racks in the pool. (See attached Figure 2-1)

QA records, during manufacture, will confirm the installation of the Boraflex poison material into every poison assembly. A neutron attenuation test will be performed at the manufacturer's plant on 10% of the storage locations after the poison assemblies are installed and locked in the racks. A site verification neutron attenuation check will be performed on at least 5 storage locations in each rack.

It should also be noted that manufacturing process Quality Assurance procedures and controls provide assurance that the actual B_4C content of the Boraflex poison material is above minimum calculational levels established by Reactor Physics calculation models for each batch of B_4C and silicone material.

A surveillance program, to demonstrate the continued integrity of the Boraflex material, will be pursued as discussed in the response to Question 3.

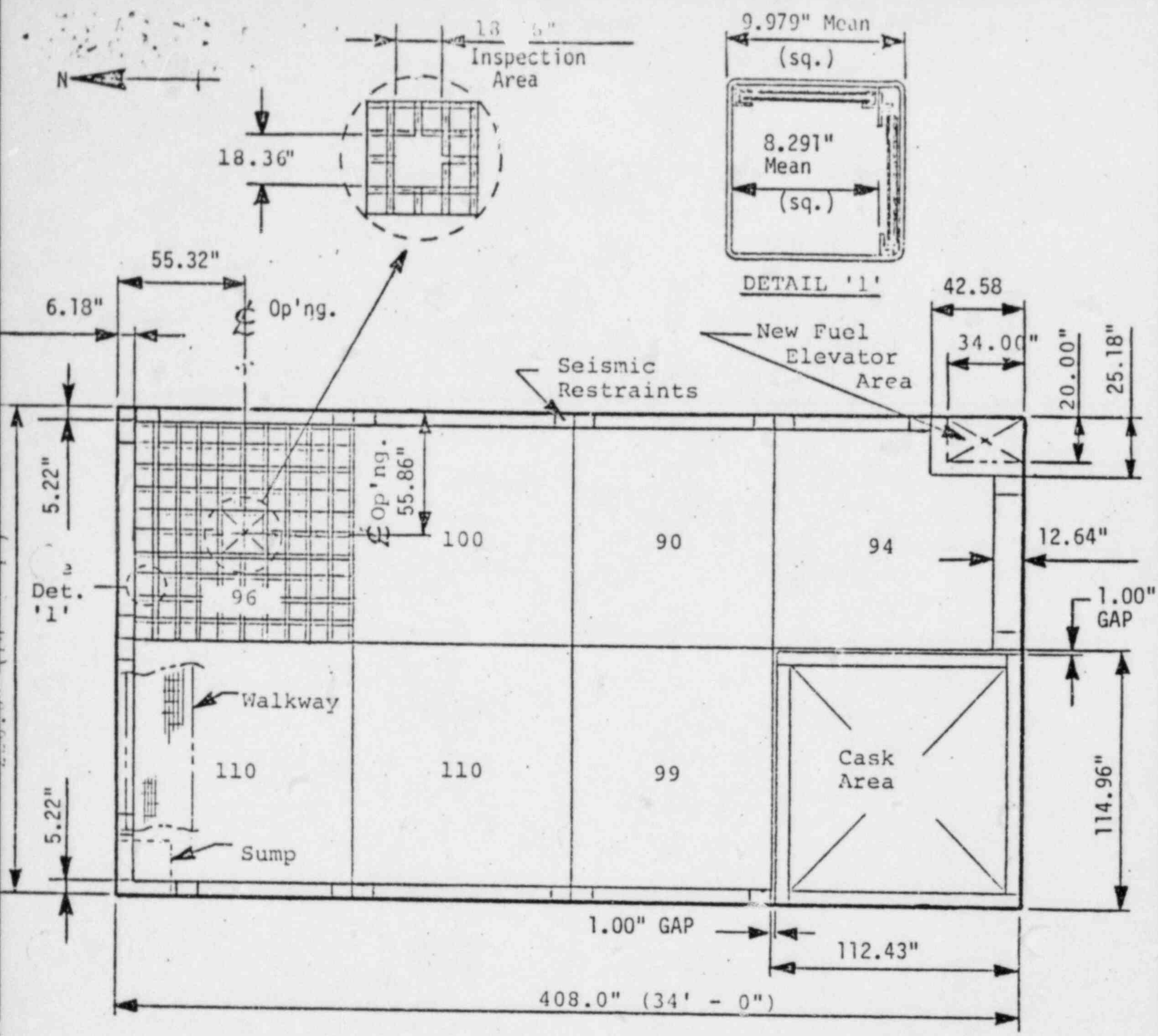
Question A-3

Describe the lifetime surveillance program that will be performed to verify the continued integrity of the Boraflex material and proper location of the boron in the storage racks.

RESPONSE

A series of 30 surveillance samples taken from the actual sheets of Boraflex installed in the racks will be encapsulated in the same manner as the rack poison and mounted on a special fixture that will fit into three outside boundary storage positions. These positions will be located in the boundary of the 99 cell rack adjacent to the cask handling area (Figure 3-1). The samples will be standard 2" x 2" squares that are used in checking the quality of the poison material as it is fabricated. These samples will be covered by 0.020 thick 304 stainless steel with the top edge of the sample vented in the same manner as the actual poison assemblies are vented. Those samples are then mounted in a picture frame 24" long x 2 1/2" wide that is suspended in the poison box location. There will be three of these assemblies maintained in the above locations.

After the first refueling three spent fuel assemblies will be placed in the cells next to the surveillance sample trains. After the second refueling one sample will be removed from each train and tested in the Penn State Research Reactor, or equivalent facility, and evaluated against previously checked control samples for reactivity comparisons. The same samples will then be decanned and tested physically and chemically for amount and distribution of boron. After each refueling a recently discharged spent fuel assembly will be moved to the surveillance sample locations thus providing maximum irradiation to the samples. Tests will be performed after two years exposure, 5 years exposure, and ten years exposure. Additional testing or the frequency of testing will be dependent upon the evaluation of the specimens removed during the early portion of the program.



PHASE I
PLAN VIEW NORTH POOL
699 STORAGE SPACES FOR
SPENT FUEL ASSEMBLIES

Figure 3-1

Question A-6

What is the weight of the largest of the proposed new rack modules?

RESPONSE

The largest rack module will weigh less than 34,000 pounds. This is calculated as follows: (Based on 304 lb./fuel box with channel inserts, lead in guides, and poison box assemblies.)

Rack Module	110 x 304 =	33,440 lb.
Rack Base Structure		300 lb.
		33,740 lb.

The base assemblies will be installed independently of the racks. Even if their weight is included the heaviest component to be lifted will be less than 17 tons.

Question A-28

Discuss the commitment of material resources required to fabricate the replacement storage racks.

RESPONSE

A total of about 220 tons of 304 stainless steel will be required for this project. In addition approximately 13 tons of silicone rubber and approximately 8 tons of B_4C will be required to complete the new racks for both pools. These values do not include manufacturing waste allowances.

None of the above material is considered critical or in short supply.