October 27, 1978

50-266

PDR

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	Docket Nos. 50-266
	)	50-301
WISCONSIN ELECTRIC POWER COMPANY	)	
	)	Amendment to License Nos. DPR-24 and DPR-27
(Point Beach Nuclear Plant, Units 1 and 2)	)	(Increase Spent Fuel Storage Capacity)

APPLICANT'S ANSWERS TO INTERROGATORIES
PROPOUNDED BY INTERVENOR
ON SEPTEMBER 27, 1978

Interrogatory 1.1

How often are gases emitted from the plant?

RESPONSE:

The plant ventilation systems operate continuously during normal operations. Releases from the spent fuel pool are vented through the drumming area vent. There are two intermittent releases of gaseous effluents - containment purges and gas decay tank releases - neither of which include releases from the spent fuel pool. A good summary description of the plant's gaseous and liquid systems is available in Appendix I of the PBNP Final Facility Design and Safety Analysis Report.

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Interrogatory 1.2

How often are emitted gases monitored for radioactivity?

RESPONSE:

All effluent releases, both liquid and gaseous, are continuously monitored for radioactivity. In addition, both gaseous and liquid effluent streams are sampled periodically for detailed laboratory analysis. For intermittent or batch releases, each batch is sampled prior to release. For continuous effluent streams, sampling is performed at least once per week for detailed laboratory analysis. Again, all these effluent streams are monitored continuously; the sampling for laboratory analyses is done to further characterize the details of the release.

Interrogatory 1.3

Have emitted gases ever exceeded the NRC allowed limits for radioactivity? If so, under what circumstances?

RESPONSE:

No. Normal plant releases range from a fraction of 1% to a few percent of applicable limits.

Interrogatory 1.4

Please quantify the predicted increase in airborne radioactivity emitted as a result of the storage of 1502 fuel assemblies.

RESPONSE:

As discussed in Section 8 of Attachment A to the application, the only two isotopes which have any potential for increasing are Kr-85 and H-3. While the inventories of these two nuclides will increase by a factor of approximately 3 if the expanded pool is filled to capacity, associated releases are expected to remain negligible. It is therefore not possible to accurately quantify the small increases, if any, in the releases of these nuclides. The following are the results of conservative bounding analyses:

(a) Less than 1% of H-3 in the storage pool originates directly from the spent fuel. As explained in Section 8 of Attachment A to the application, most of the H-3 in the spent fuel pool originates from other plant operations unrelated to the number of assemblies stored in the pool. The drumming area vent exhausts ventilation air from the spent fuel pool area, the waste packaging area, and a portion of the auxiliary building; for this analysis, all tritium releases through the drumming area vent is assumed to originate from the spent fuel pool. With these grossly conservative assumptions and based on current releases through the drumming area vent, an increase of about 4 Curies is calculated. This release would result in a maximum dose of 0.00028 mrem/year to an individual living near the site boundary. The actual increase will be less, probably substantially less.

(b) For Kr-85, the analysis is also ridiculously conservative. All the gases observed through the drumming area vent are assumed to originate from the spent fuel. Based on some observed data, about 80% of Drumming Area Vent releases consist of Xe-133. For this analysis, the remaining 20% is

conservatively assumed to be all Kr-85. With these assumptions, an increase of about 150 Ci is calculated. This release would result in a maximum dose of 0.000031 mrem/year to an individual living near the site boundary. Again, this is a bounding analysis, not an estimate. Actual releases, if any, are expected to be substantially less.

As a practical matter, there is essentially no release of radioactivity from spent fuel assemblies after the first few months when the temperature has been reduced to the stage where there is no longer a substantial temperature differential between the fuel rods and the pool water to drive nuclides out of the rods. Since all spent fuel assemblies would reside in the pool during this period, regardless of the ultimate capacity of the pool, i.e., whether or not new racks are installed, the increased number of fuel assemblies ultimately residing in the pool would have an insignificant effect on the release of airborne effluents from the pool. Thus, the incremental number of curies calculated above would not be expected to actually be released.

Interrogatory 1.5

Does the State of Wisconsin monitor the Point Beach emissions? Where, when and how?

RESPONSE:

The Section of Radiation Protection of the Department of Health and Social Services of the State of Wisconsin routinely conducts a radiological environmental monitoring program in the vicinity of Point Beach Nuclear Plant. A copy of a recent report titled, "Annual Report, Environmental Radiation Survey, Point Beach-Kewaunee Sites, January - December 1976" prepared by the Section of Radiation Protection, Division of Health, Wisconsin Department of Health and Social Services is being provided to the intervenors. Sample locations, types, analyses, and frequencies are indicated therein.

Interrogatory 2.1

Could the total airborne radioactive emissions of the combined releases from Point Beach's and Kewaunee's Nuclear Plants with expanded fuel storage exceed the NRC's limits to residents living between the plants? Please provide data leading to your conclusions.

RESPONSE:

No. Cumulative effects are negligible. To place this question in perspective, consider a very conservative bounding case: two Point Beach sites immediately adjacent to each other such that the south boundary of "Point Beach North" coincides with the north boundary of "Point Beach South". Doses at the coincident boundary are calculated applying known meteorology for north and south sectors. Assuming the releases for H-3 and Kr-85 as given in the response to Interrogatory 1.4, the doses are 0.000038 mrem per year from Kr-85 and 0.00035 mrem per year from H-3. This represents increases of 0.000007 and 0.00007 mrem per year, respectively, as compared with the single plant doses presented in the response to Interrogatory 1.4. While these doses are already negligible, it is important to note that the releases assumed are grossly conservative, the distance between one plant and the other's site boundary is about 3.5 miles for the Point Beach-Kewaunee situation, and spent fuel storage at Kewaunee is less than at Point Beach. Hence the actual cumulative effects will be even less.

Interrogatory 2.2

Would a local thermal inversion cause the combined releases to remain in the area and pose a hazard?

RESPONSE:

No. The combined releases result in negligible effects, and a local thermal inversion would not pose a hazard. In any event, for this site, the occurrence of lake and land breezes together with the absence of unusual geographical features virtually preclude the meteorological condition suggested in the question. Furthermore, the meteorological analysis accompanying the Appendix I dose evaluation for Point Beach was performed in accordance with NRC accepted procedures and already incorporates terrain correction factors which conservatively account for air flow reversal phenomena which are inclusive of the type of meteorological event postulated in the question. In general, the atmosphere is unable to maintain such a hypothetical state of motion for an extended time period.

Interrogatory 2.3

Provide record of emissions for previous year.

RESPONSE:

Copies of the two Semiannual Monitoring Reports for Point Beach Nuclear Plant for calendar year 1977 are being provided to the intervenors.

Interrogatory 3a.1

Predict the maximum temperature and heat load to the spent fuel pool in the event both cores must be unloaded into the filled spent fuel pool. Discuss the result of pump failures in this situation.

RESPONSE:

With a spent fuel pool storage capacity of 1502 assemblies, the most conservative assumptions is to have 1260 storage positions filled, with 242 spaces available for unloading the two cores. Two core unloads in this situation is not realistic but is evaluated for purposes of this interrogatory.

In order to evaluate the pool water temperature, it is first necessary to establish a time sequence of events and calculate the total heat load in the pool. The computer program identified in Section 4.2 of Attachment A to the application has been utilized to develop the heat loads for this situation. Figure 1 attached is a plot of the decay heat for a full core unload as a function of time after reactor shutdown.

The heat load that would be in the spent fuel pool is established as follows, assuming that thirteen days are required to unload a core and that the cores are unloaded sequentially with one day pause between unloadings:

a.	decay heat from second core unload 13 days after shutdown of reactor from Figure 1.	11.5×10^6 BTUs/hr
b.	remaining decay heat from first core unload 27 days after shutdown of reactor (13 days for unloading the first reactor, 1 day pause, 13 days for unloading the second reactor; from Figure 1).	8.3×10^6 BTUs/hr
c.	residual decay heat from 1260 in-storage assemblies (use 1280 assembly line from Table 4-1 of Attachment A to the application - no further decay accounted for)	9.38×10^6 BTUs/hr
	Total decay heat load in pool	29.18×10^6 BTUs/hr

Normally, only one of the two cooling trains is used to maintain the temperature of the pool water at 120°F or less. Using both trains, the cooling system has a design capability to maintain the pool temperature at 120°F with a heat load of 28.2×10^6 BTUs/hr. The above calculated number exceeds the design capability of the cooling system by 0.98×10^6 BTUs/hr, or less than 3.5%. However, there is over 5% more heat transfer surface area in each heat exchanger (per the heat exchanger technical manual data sheet) than is used to calculate the design capability. Thus, the cooling system could accommodate the above postulated heat load and still maintain the pool temperature around 120°F.

Section 4.3 of Attachment A to the application evaluates the capability of one cooling train accommodating an abnormally high heat load. The submittal states that with a heat load of 23.9×10^6 BTUs/hr, one cooling train can maintain the pool temperatures below 145°F. In answering this interrogatory, both nuclear cores must assume to be off-loaded into the spent fuel pool for some reason, and then one cooling train is postulated to fail.

An accident analysis procedurally identical to that of Section 4.3 of Attachment A to the application shows that a single cooling train could accommodate a heat load of 29.2×10^6 BTUs/hr, and maintain the spent fuel pool temperature at about 162°F. As the heat load would decay away with time, the maximum pool temperature would also be reduced.

The temperature results of a complete loss of cooling is strictly related to the thermal inertia of the spent fuel pool. Given that the spent fuel pool contains about 43,700 cubic feet of water, that the specific heat of water is 1.0 BTU/lb_m-°f, that the water density would average about 61.0 lb_m/ft^3, then the thermal inertia can be calculated as follows:

$$\text{Thermal Inertia} = 43{,}700 \text{ cu. ft.} \times \frac{61 \text{ lb}_m}{\text{cu.ft.}} \times \frac{1 \text{ BTU}}{\text{lb}_m\text{-}^\circ\text{F}}$$

$$= 2{,}665{,}700 \text{ BTUs}/^\circ\text{F}$$

Thus, about 2.66×10^6 BTUs is required to raise the pool water temperature 1°F.

To calculate the time required to raise the pool water temperature, it is necessary to postulate the heat load existing when cooling is lost (q), the temperature existing when cooling is lost (T initial), and the temperature level to which the pool water will be allowed to rise (T final). Then the equation is simply

$$\text{Time} = \frac{\text{Thermal Inertia} \times \text{temperature increase}}{\text{heat load}}$$

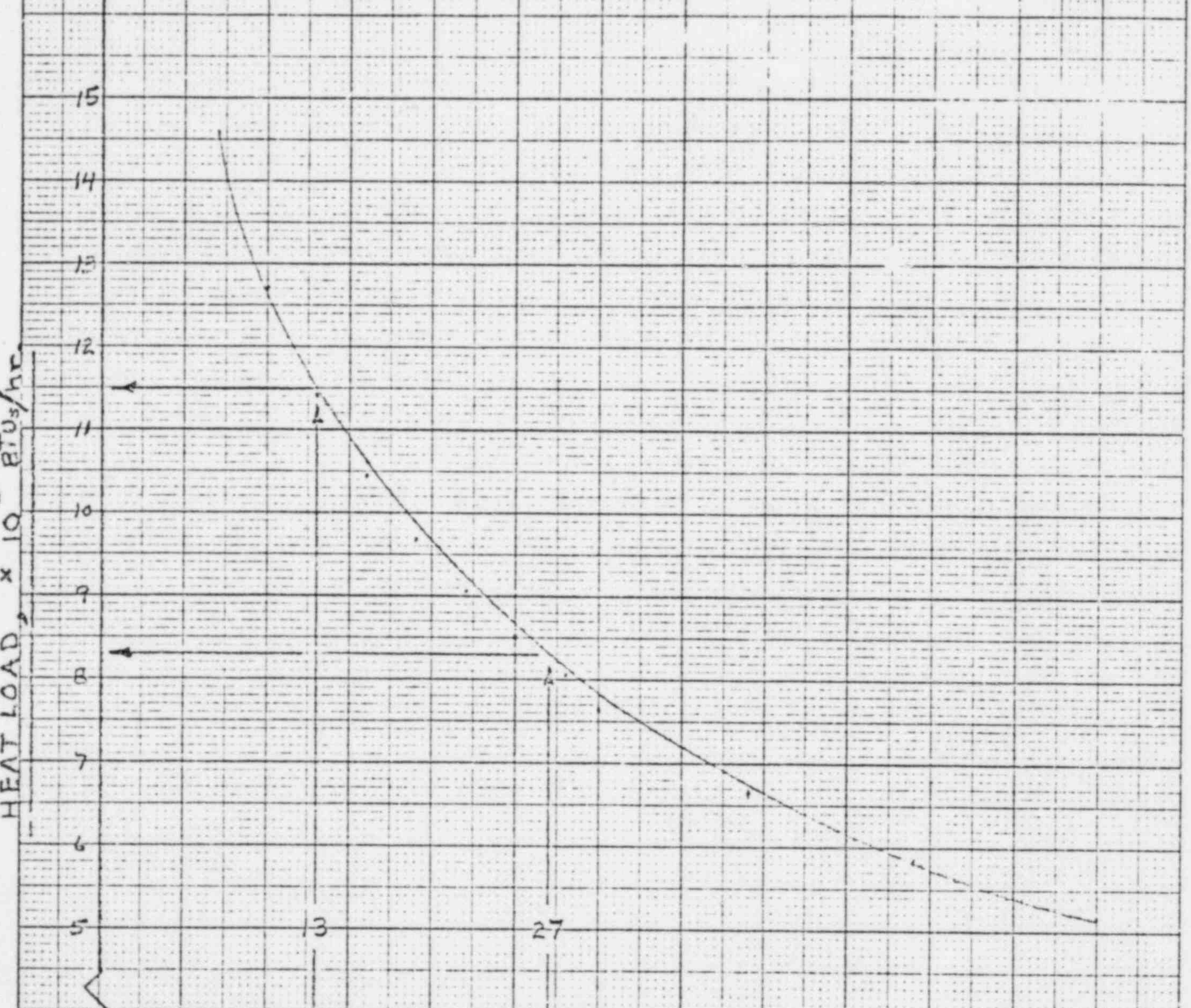
The time required to heat up the PBNP spent fuel pool for various heat loads at three different temperature increases is shown in Figure 2.

The results in Figure 2 are conservative because no credit is ta' ו for evaporative cooling from the pool surface or for the heat capacity of other items in contact with the pool water (the pool steel liner, storage racks, and the metal in the stored spent fuel assemblies). Thus, the actual time required to heat up the volume of pool water would be longer than the calculated value.

Figure 1

DECAY HEAT vs. TIME FOR PBNP FULL-CORE UNLOAD

Based on A. 121 fuel assemblies
B. 3.2 % enriched uranium fuel
C. avg. burnup of 22,000 MWD/MTU
D. based upon calculated values



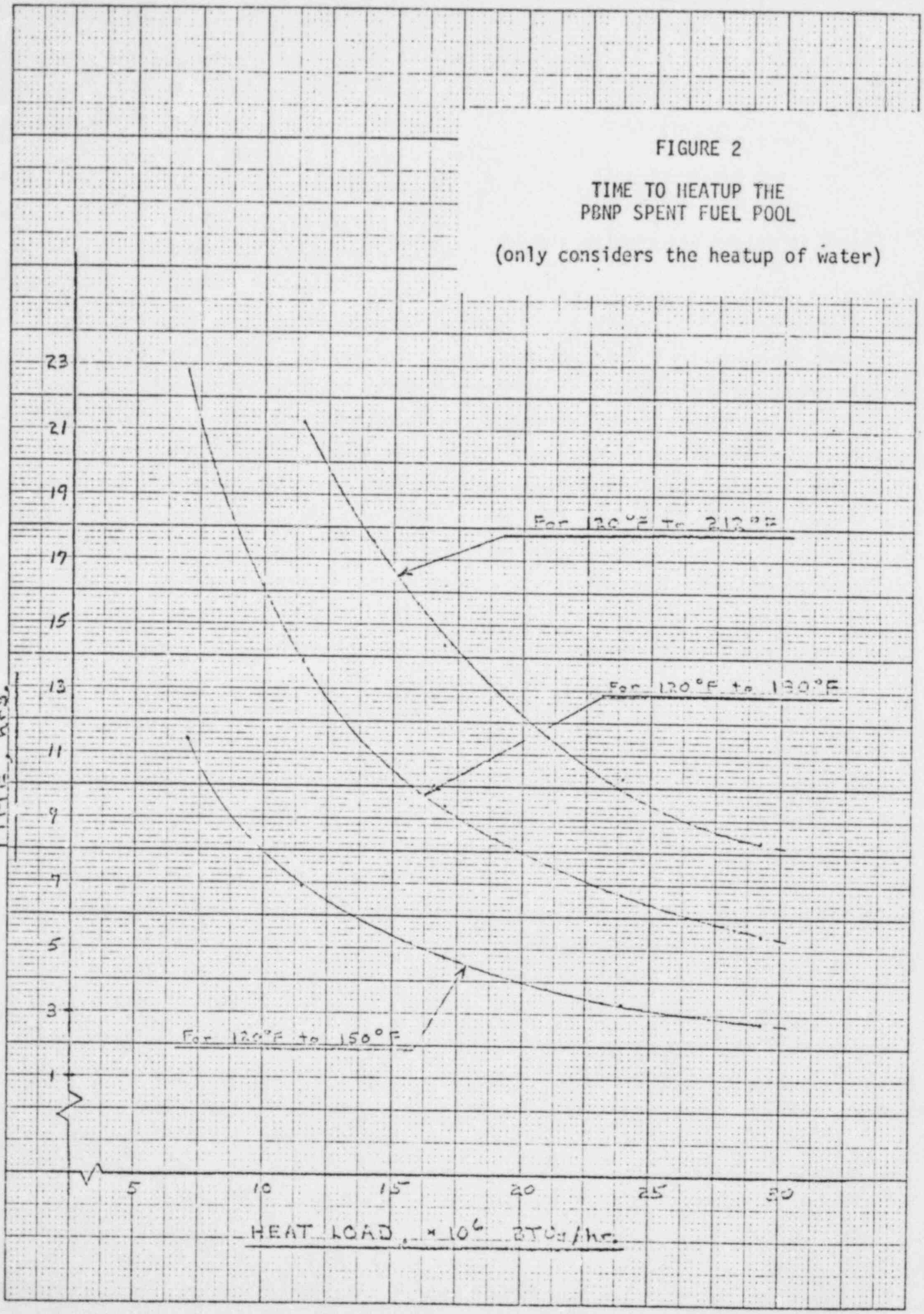


FIGURE 2

TIME TO HEATUP THE
PBNP SPENT FUEL POOL

(only considers the heatup of water)

Interrogatory 3b.1

Will any spent fuel assembly, after being racked following removal from the core, ever be moved to a different storage location prior to being removed for shipment?

RESPONSE:

Yes, it is possible that a spent fuel assembly, after being placed in a spent fuel rack following removal from the core, may be moved to a different storage location prior to being removed for shipment. Shortly after the fuel assemblies have been removed from the core and placed in their storage locations, a fuel assembly will be taken from its storage location and moved to an inspection location for its detailed visual inspection. After this inspection, it is returned to its initial storage location. Generally all fuel assemblies removed from the core during a refueling will be inspected within a month after the refueling.

After this initial inspection and return to the storage location, it is unlikely that a fuel assembly will be moved to another storage location prior to shipment. Rarely there may be a need to relocate several fuel assemblies in order to accommodate a subsequent refueling region unloading. However, the utilization of storage locations in the spent fuel pool is planned in advance to minimize, and, as much as possible, eliminate the necessity for relocating spent fuel assemblies to other storage locations.

Interrogatory 3b.2

Does moving a spent fuel assembly from one storage location to another require that workmen be positioned above the spent fuel pool?

RESPONSE:

Yes. Spent fuel assemblies are handled in the spent fuel pool by a long-handled tool suspended from an electric monorail overhead hoist which is positioned on a wheel mounted walkway which spans the spent fuel pool. The hoist travel and tool length are designated to limit the maximum lift of a fuel assembly to a safe shielding depth. The overhead hoist and handling tool are manipulated by an operator standing on the moveable bridge over the pool.

Interrogatory 3b.3

What measures are taken to prevent articles (tools, clothing) associated with workers' activities above the pool from falling into the pool?

RESPONSE:

Several precautions are taken to prevent workers' activities from causing articles to fall into the spent fuel pool. As a general rule, all personnel entering the controlled zone, which includes the spent fuel pool area, are required to wear protective clothing. Generally, this consists of coveralls or laboratory coats, shoe covers, gloves, and surgeon's caps. To work in the area immediately adjacent to the spent fuel pool or on the spent fuel pool moveable bridge, a Radiation Work Permit must be issued. This is a specific permit for a designated area and activity. Before working on the moveable walkway bridge over the spent fuel pool, workers must take the precaution of taping shut all coverall or lab coat pockets that contain items which could fall out and into the pool. Individuals who wear glasses must wear a head band to prevent the glasses from slipping off into the pool. All tools which are to be used over the spent fuel must be tethered to the bridge so that if they are accidentally dropped, they cannot fall into the pool.

Interrogatory 3b.4

Does the spent fuel pool's coolant water suction pipe have a protective screen to prevent the drawing of items dropped into the pool?

RESPONSE:

No. The spent fuel pool coolant suction pipe is located in the northwest corner of the north half of the spent fuel pool. The suction pipe enters the pool water vertically, from above, and is terminated three feet below the normal water surface elevation. Thus, the end of the pipe points downward with an open horizontal plane located below the water surface.

Interrogatory 3b.5

What would be the result of a blockage of the suction pipe of the spent fuel pool's coolant system?

RESPONSE:

Any items that would fall into the spent fuel pool would either float on the water surface or sink to the bottom of the spent fuel pool. As items floating on the water surface are three feet above the pipe opening, it is inconceivable that anything could get sucked down and then up into the pool suction pipe.

The pool suction pipe is nominal ten-inch diameter (10.020" inside diameter) piping. Because the suction pipe feeds two cooling trains, it is reduced to nominal eight inch diameter (7.981" inside diameter) piping prior to reaching each pump. If somehow the eight inch pipe was blocked, the cooling could simply be transferred to the other cooling train. If somehow the ten inch pipe was blocked, the cooling system would have to be shut down until the pipe was cleared. There is a set of bolted flanges in the ten inch piping (originally installed for pressure testing purposes) and bolted flanges are used to connect the piping to the pumps. These flanges could be disconnected and the lines cleaned out if necessary.

Interrogatory 3b.6

Will increased number of stored assemblies require greater worker activity to monitor them?

RESPONSE:

No. An increase in the number of spent fuel assemblies stored in the spent fuel pool will neither necessitate nor result in greater worker activity to monitor the assemblies. As discussed in the response to Interrogatory 3b.1, once a fuel assembly has been removed from the reactor core and inspected, it is unlikely that the assembly will be moved from its storage location until it is shipped. Such storage pool parameters as water temperature, pool water level, water chemistry and operation of the spent fuel pool cooling system are periodically monitored by plant personnel. This monitoring, however, is independent of the number of fuel assemblies stored in the pool.

Interrogatory 3c.1

What would be the maximum spent fuel pool coolant water temperature if the pool's racks were filled to capacity (except spaces left open for emergency core offload needs) and both coolant pumps failed?

RESPONSE:

A core off-load requires space for 121 assemblies. Thus, the postulated scenario has 1381 spent assemblies stored in the pool. Assuming that the failure occurs immediately following a refueling, the heat load that would be in the pool would be about 10.4×10^6 BTUs/hr (Applicant's submittal, dated March 21, 1978, Attachment A, Table 4-1, second line from the bottom). This heat load could easily be accommodated by one cooling train while maintaining the pool water temperature below 120°F.

Assuming both pumps fail and that one of the two cooling trains can be placed back into operation in less than five hours, then the maximum pool water temperature would be less than 150°F. This is shown in Figure 2 of the response to Interrogatory 3a.1. The resulting temperature for other time periods for placing one cooling train back into operation can also be obtained from Figure 2.

The simultaneous failure of both cooling trains is not considered credible. The cooling system has been seismically designed so that earthquake forces will not mechanically affect the system. The cooling system and components are all located substantially away from any high energy piping. Thus, the postulated failure of these piping systems would not affect the cooling system. The pump motors of the two cooling trains are powered from separate motor control centers. Thus, the failure of one motor control center would only affect one of the cooling trains. In addition, these motor control centers can be individually powered by the two in-plant emergency diesel generators should electrical power from all off-site sources be lost.

Interrogatory 3c.2

Given the possibility that both pumps fail while the spent fuel pool is holding the maximum number of spent fuel assemblies (assume both cores are emergency off-loaded) would be spent fuel pool's coolant water boil?

RESPONSE:

Assuming that the cooling system was working and holding the pool temperature at about 120°F before it failed, and assuming that at least one cooling train could be fixed and made operational in less than five hours, the pool water would not boil. For other assumed time periods, Figure 2 in the response to Interrogatory 3a.1 shows the resulting pool temperature when the cooling train is placed back into operation.

Interrogatory 3c.3

What is the "Design Basis" maximum water temperature allowed to circulate through the spent fuel pool coolant pumps?

RESPONSE:

The spent fuel pool pumps have a design temperature of 200°F.

Interrogatory 3c.4

If due to pump failure, emergency core offloads and increased stored fuel assemblies the spent fuel pool's coolant water should boil, what would be the results in air-borne radioactive emissions released from the plant?

RESPONSE:

The tritium in the amount of pool water boiled away would be released. If 1% of the pool volume were to be lost by the boiling, for example, the release would be about 0.4 Curies. There would be no effect on the fuel itself, since it is designed to withstand reactor temperatures in excess of 600°F, far greater than the temperature of water boiling in an open pool. There would be no significant increase in the release of other nuclides.

Interrogatory 3d.1

If through accidental breach or corrosion, the spent fuel pool should lose volume of coolant water, what would be the results in airborne radioactive emissions from the spent fuel pool?

RESPONSE:

The loss of large volumes of water from the spent fuel pool is not considered credible because of the design considerations of the pool. The design is such that water will always be covering spent fuel assemblies stored in the pool and this water acts as the medium for removal of decay heat from the fuel assemblies. This requirement for coverage of fuel with water exists for the present pool configuration and it will be required after the proposed rerack as well.

In the event of a leak, the pool water inventory can easily be maintained by adding water equal to the rate of leakage until the liner is repaired. Adding water to the spent fuel pool can be accomplished by many means. When the new cooling system was installed, an emergency cooling water makeup connection was included in the seismically designed service water supply piping. This connection was installed simply to provide a source of water for the spent fuel pool if required in an emergency. Water from this source can be added to the pool at a rate of 250 gpm for an indefinite time period. Some of the other sources for makeup water and their delivery capacities are as follows: reactor makeup water - 200 gpm for 13 hours, refueling water storage tank - 100 gpm for 45 hours, water treatment plant - 85 gpm indefinitely, fire water system - 1000 gpm indefinitely.

We interpret this interrogatory to be asking for the airborne radioactive emissions should the loss of pool water uncover the spent fuel assemblies stored in the pool. As explained above, we consider such an event to be incredible and therefore, no calculation of this type has been performed. If one were to perform such a calculation to examine the incremental effect due to the proposed rerack, there would be no significant incremental effect. This is due to the fact that the rerack allows

for storage of fuel that is older than four years of age and the major source of heat generation and radiation is the freshly discharged fuel from the reactors. This discharge has and will occur twice a year independent of any reracking.

Interrogatory 3d.2

What would be the thermal effects on the stored fuel assemblies, cladding and racks at a partial or total loss of coolant water from the spent fuel pool?

RESPONSE:

As noted in the response to Interrogatory 3d.1, the total loss or large loss of water from the spent fuel pool is considered incredible. Therefore, no analysis of the thermal effects on the stored fuel assemblies, cladding and racks was performed under these conditions.

Interrogatory 3e.1

If through accident or corrosion, the pool's liner should leak while the spent fuel pool is filled to capacity:

a. where would the stored spent fuel assemblies be kept during repairs?

b. how would repairs be made?

RESPONSE:

Should a leak occur to the liner at some time in the future, the spent fuel will be stored in the pool in storage locations as remote from the leak as is possible. One empty rack module, with 110 storage locations or less, could be removed to provide access to the area of the leak for repair.

For the remote case when the pool is completely filled, two options exist: ship off-site to another pool for temporary storage 110 fuel assemblies, or place a rack in the cask handling area for temporary storage of fuel assemblies. Both options would allow fuel to be removed from the rack in the area of the leak and the rack to be removed to provide access for repairing the leak.

The repair procedure for repairing a leak in the pool liner would depend on the location and severity of the leak. A leak above the minimum water level over the top of the fuel could be repaired by dewatering to the level of the leak and weld repairing in the dry condition. If a leak were identified below the minimum water level, it could be repaired by welding using a diver. Diving work in fuel pools has been performed at Point Beach and other sites in the past. Underwater welding has also been performed on stainless steel fuel pool liners similar to that of Point Beach.

If it was desired to avoid underwater diving work, a leak located on the bottom of the pool or below the minimum water level could be repaired by working inside an evacuated chamber, such as a large diameter pipe caisson. The caisson would be

jacked against the liner, with a gasket on its leading edge, and the water pumped out. It might be necessary to remove a fuel rack to get at the damaged area. Single fuel racks can be removed without removing adjacent racks.

Interrogatory 3f.1

Please present data for conclusions in Sec. 5.0 of the spent fuel storage modification description.

RESPONSE:

The pool structure, including the pile foundation, was analyzed to determine the adequacy of all structural components under the revised loading condition. For the analysis, assumed rack and fuel loads were used because the analysis was started prior to completion of the final rack layout and design. The assumed loads are higher than the actual loads that result from the final design, so the results of the analysis are conservative in all cases.

The data used for concluding that the piles are adequate to support the additional rack and fuel load was previously provided in response to NRC Question C-18, submitted via letter dated October 10, 1978.

The fuel pool slab and walls were evaluated using load combinations and factored loads as described in Section 5.0 of the Applicant's amended submittal of March, 1978, and the response to NRC Question C-8 submitted via letter dated October 10, 1978. The results of the analysis (using the assumed loads which are higher than the actual loads) are as listed in Table 3f-1.

TABLE 3f-1

ANALYSIS DATA - MAXIMUM ALLOWABLE LOAD VS. ASSUMED LOAD

Portion of Structure	Maximum Allowable Load	Assumed Load
Base Slab	Mu 286	261
	Vu 72.4	36.4
West Wall	Mu 246	220
	Vu 63.4	57.5
North & South Walls	Mu 246	192.5
	Vu 63.4	49.9
East Wall	Mu -394	-242
	Vu 60.9	60.9

Mu = moment load, ft-K/ft
Vu = shear load, K/ft

The data presented is for the most highly stressed portion of that component of the structure. Other areas have less critical loading than those listed above, using the factored load equations. The actual loads do not exceed the allowable at any location, so it is concluded that the structure is adequate to support the new rack and fuel loads.

Interrogatory 3f.2

How will the discovery of the Lake Michigan fault line near Point Beach affect the safety of the spent fuel pool? Please present data.

RESPONSE:

Category I structures at Point Beach, including the spent fuel pool, are designed to withstand the maximum earthquake hypothesized for the site. The design earthquake envelopes the magnitude of the largest earthquake that has occurred within 200 miles of the site. It is also larger than the effect at the site of any earthquake in recorded history that has occurred greater than 200 miles from the site.

The 200-mile radius discussed above includes the possibility of faulting in Lake Michigan. A study of the seismicity of Lake Michigan and the surrounding area was undertaken to evaluate the capability of recently postulated faults beneath the lake[1]. It was concluded, based on this study, that there was no evidence of earthquake activity originating in Lake Michigan; consequently, the postulated faults are not considered capable, and therefore would not affect the safety of the spent fuel pool at Point Beach.

(1) "Geology and Seismicity Under Lake Michigan" Appendix 2M Haven Nuclear Plant Units 1 and 2 Site Addendum, Preliminary Safety Analysis Report Volume [illegible].

Interrogatory 3f.3

To what Richter Scale seismic event is the spent fuel pool built to withstand?

RESPONSE:

The earthquake magnitude used for the Point Beach design is 0.12g's in the horizontal direction and 0.08g's in the vertical direction. This corresponds to a magnitude of about 5.5 on the Richter Scale, and about VII on the Modified Mercalli Scale.

Interrogatory 3g.1

Will the present storage racks be removed and shipped offsite intact, or will they be cut up and shipped in barrels?

RESPONSE:

The final decision of shipping in toto vs. cutting up has not yet been made. Please refer to the answer to NRC Question A-29, submittal dated July 19, 1978, for detailed discussion on this subject. Note that in the event racks are to be cut up, shipping would not be in drums but in crates.

Interrogatory 3g.2

What is the maximum allowable radiation a worker may receive in a year? In a calendar quarter?

RESPONSE:

By 10 CFR §20.101, exposure of the whole body is limited to 3 Rems per calendar quarter. The lifetime cumulative dose is limited to 5(N-18) Rems where N is the individuals' age in years. In other words, an average of 5 Rems per year is permitted for each year after age 18. If an individual's annual average is below 5 Rem he can receive an additional amount not exceeding the 3 Rem per quarter limit; in other words, the maximum allowable dose under any circumstances is 12 Rem per year.

Interrogatory 3h.1

Submit results of experiments on the deterioration of neutron absorber plates presently being carried out by the University of Michigan. Discuss the effects of high level gamma radiation damage and gas generation in the B_4C plate.

RESPONSE:

A summary of the results of previous testing on the Boraflex poison material is contained in an eleven page Wisconsin Electric, Nuclear Projects Office memorandum of June 26, 1978; see copy attached hereto. The estimated gamma radiation exposures contained within this memorandum were preliminary numbers; the correct numbers are as presented in the October 10, 1978, response to NRC Question C-2. A copy of the BISCO Report 1047-1 is being provided to intervenors.

Additional testing is planned to commence at the University of Michigan on or about October 16, 1978. Samples in three different environments will be exposed to various levels of gamma radiation. The environments will be air, deionized water, and deionized water with 2000 ppm boron in the form of boric acid (each sample will be in its own container). Control samples will also be maintained in corresponding environments so that the relative effects can be determined.

June 26, 1978

Mr. T. R. Wilson/File 4.9.5

POISON MATERIAL FOR THE
PBNP SPENT FUEL STORAGE RACKS

Because of the recent problems experienced with the Connecticut Yankee plant spent fuel storage racks (off-gassing of the poison material with attendant bulging of the encapsulating steel and subsequent stuck fuel assemblies), this memorandum is written to compare pertinent poison material parameters and to summarize the known poison material testing results for the Point Beach high density spent fuel racks.

The poison material to be used in the PBNP spent fuel racks is called Boraflex and is a silicone rubber, boron carbide compound with a minimum B_4C loading of 34.8% by weight. The material is fabricated by Brand Industrial Services, Inc. (BISCO) who has prepared a report (No. 1047-1) that presents the results of the testing already conducted on the Boraflex material. One copy of this report is in the Nuclear Projects Office; Attachment A hereto is based upon the BISCO report.

In addition to the testing that has already been concluded, Wachter Associates has advised that additional tests are being performed. Because of the off-gassing situation, both at Connecticut Yankee and as noted in the test reports, the fabrication process for the poison material has been changed to include oven-drying of the boron carbide material and oven-curing of the formulated Boraflex material. All of the testing is, or will be, repeated with the oven-cured material. Also the high temperature soak tests in borated water (see Attachment A, item 4) are continued and have accumulated about 280 days of testing to date.

The problem that occurred at Connecticut Yankee (stuck fuel assemblies) should not develop at Point Beach because of a basic design difference: the poison material in the Point Beach racks will be contained within a tight-fitting stainless steel "bucket" open to the water where the Connecticut Yankee poison material was completely enclosed. Thus, generated gas will be able to escape rather than bulge the poison material container. Table 1 summarizes some of the differences between the Connecticut Yankee and Point Beach poison materials and storage racks.

To further evaluate the acceptability of the Boraflex poison material, the following parameters are presented. The radiation dosages are based upon the following cases; "fresh" - where every six months a recently discharged spent fuel assembly is placed in the same position with the previously installed assembly being relocated and "three equal" - where a spent fuel assembly is stored for about 13 years in the same position and then replaced with a recently discharged spent assembly.

a.	Conservatively estimated gamma radiation exposure, rads	1.5×10^{12} 6×10^{11}	"fresh" "three equal"
b.	Water temperature around poison (Calc. 128S10, Pg. 31), °F	170 less than 240	expected worst case
c.	Poison material temperature (Calc. 128S13, Pg. 5), °F	178 248	expected worst case
d.	North pool exit water temperature (with cooling - Calc. 128S10, Pg. 31)	150°F	worst case
e.	Approx. surface area of a poison slab (0.1 in. x 8.5 in. x 145.5 in.)	2502.6	in^2
f.	Estimated total poison material surface area in pool	7,087,363 (2832 of "e")	in^2

Attachment A test results show that the material releases gas due to both irradiation and exposure to high-temperature borated water. Because the testing was performed individually, the combined effects are not known. Also, during the tests the Boraflex was not covered with stainless steel sheets and thus this effect on the gas release rate is not known, and the spent fuel pool will be at a much lower temperature and have a lower boron content and their effects are not known.

However, if it is assumed that the data of Attachment A, item 4, is applicable (5 in^3 of gas/in^2 of surface area with 35% of the gas released in the first 25 days following installation), the following would result:

South Pool

a. 803 storage positions with about 1500 poison slabs

b. $2500 \, \frac{in^2}{slab} \times 1500 \text{ slabs} \times 5.0 \, \frac{in^3 \text{ of gas}}{in^2 \text{ of slab area}} = 18{,}750 \times 10^3 \, in^3 \text{ of gas}$

c. $18{,}750 \times 10^3 \, in^3 \text{ of gas} \times \frac{1 \, f^3}{1728 \, in^3} = 10.85 \times 10^3 \, f^3 \text{ of gas}$

d. $\frac{10.85 \times 10^3 \, f^3 \text{ of gas}}{25 \text{ days}} \times \frac{1 \text{ day}}{24 \text{ hrs}} \times \frac{1 \text{ hr}}{60 \text{ min.}} = 0.301 \text{ cfm}$

A gas release rate of 0.3 cfm (one 8-inch cube a minute) is not very significant with respect to gas volume. While the North pool is to be reracked first, the North pool will contain less poison slabs and therefore the gas release rate would be smaller for the North pool.

The estimated gamma radiation to which the poison material would be exposed during 40 years has been conservatively estimated at 1.5×10^{12} rads. It must be noted that this exposure is greater than the reported testing exposure and thus the anticipated Boraflex behavior is not known. The Cobalt 60 testing showed that at an exposure of 7×10^{8} rads gamma the material had become quite brittle. Because of the high temperature (about 300°F), the results are in question but it should be noted that the thermal aging tests (at 350°F) did not produce embrittlement to the extent that the gamma radiation exposure showed and thus the gamma radiation is concluded to be a major affect.

Based upon the testing results (Attachment A) and the analysis of the Point Beach racks, the following conclusions can be reached concerning the Boraflex poison material.

1. The material will become embrittled due to the gamma radiation but because it is contained within a bucket (see NRC submittal, Attachment B, Page 2-4) it will be retained in place.

2. The material is acceptable in a boron water environment with dimensions decreasing.

3. Off-gassing will occur probably for an extended period of time but not at a very large release rate.

ORIGINAL SIGNED BY
D. L. DILL
D. L. Dill

/ldk

Attachment

Copies to Messrs. Sol Burstein w/attachment
G. A. Reed w/attachment
File 4.9.5 w/attachment

COMPARISON OF SPENT FUEL RACK POISON MATERIALS

	Parameter	Connecticut Yankee[1]	Point Beach
1.	Type of poison	B_4C plates with binder	B_4C in a silicone rubber
2.	Manufacturer -	Carborundum Co.[2]	BISCO
3.	Completely encapsulated?	Yes	No
4.	Previously tested?	Yes	Yes
	a) irradiation	2×10^{11} rads by electron beam	8.5×10^{17} neutron 7×10^{9} rads gamma
	b) in water	Yes	Yes, with 3000 ppm boron
	c) thermal cycling	to 350°F	Yes, at 240°F
	d) off-gassing constituents and percentage	H_2 - 18% by vol.	40.9
		O_2 - 3% by vol.	6
		CO_2 - 8% by vol.	-
		N_2 - 69% by vol.	33.7
		CH_4 - 1% by vol.	19.5
5.	Apparent Min. Gap between FA size and storage position, inch	0.190	~0.480
6.	Racks installed	Summer 1977	Summer 1979

(1) All data from Licensee Event Report CTHNP1, 78-004/01 T 0, dated 5/12/78.

(2) Also was the supplier for the Kewaunee storage racks poison material. Would not tell WPS the binder composition but since the development of the problem it is believed that the binder composition has been provided to NRC. WPS has cancelled purchase order since the development of the Connecticut Yankee problem and is now working with a company in Germany.

ATTACHMENT A

SUMMARY OF BISCO REPORT 1047-1

The subject report (Revision 1 dated May 5, 1978) is entitled "Boraflex 1 Suitability Report" and is compiled and published by Brand Industrial Service Inc. of Park Ridge, Illinois. The report is in a loose-leaf 3-ring binder, and is about one inch thick. The report includes various summaries, data sheets, and BISCO promotional literature. The following excerpts are taken from various portions of the report.

1. Thermal Aging Tests

These tests were performed in a controlled temperature oven at 177°C (about 350°F) and at 190°C (about 375°F). Tests of physical properties were conducted at various times during the thermal aging testing periods which were about 245 days and 210 days respectively. The results were as follows:

Time, hrs.	Durometer		Tensile		Elongation	
	@177	@190	@177	@190	@177	@190
7 days @ RT	53	53	460	460	116	116
240	63	63	549	444	78	84
480	59	62	404	397	84	90
960	64	62	490	364	78	92
1920	62	63	404	353	80	83
2880	63	-	275	-	60	-
4080	62	65	271	263	64	70
5160	62	64	267	247	68	70
5880	63	-	278	-	57	-

From the above data, at 177°C (350°F) the property changes seem to have stabilized after about 3000 hours (about 125 days). The testing at 190°C does not appear to have been long enough to stabilize the properties.

2. Effects of Gamma Radiation from a Cobalt 60 Source

The data is presented on page 4. The samples were not cooled during the testing and it is estimated that the temperature reached about 300°F. The data shows that gamma irradiation makes the material brittle.

3. Irradiation Testing of 50% B_4C Boraflex

The samples were irradiated and tested at the University of Michigan. The long term irradiation data is presented on page 5.

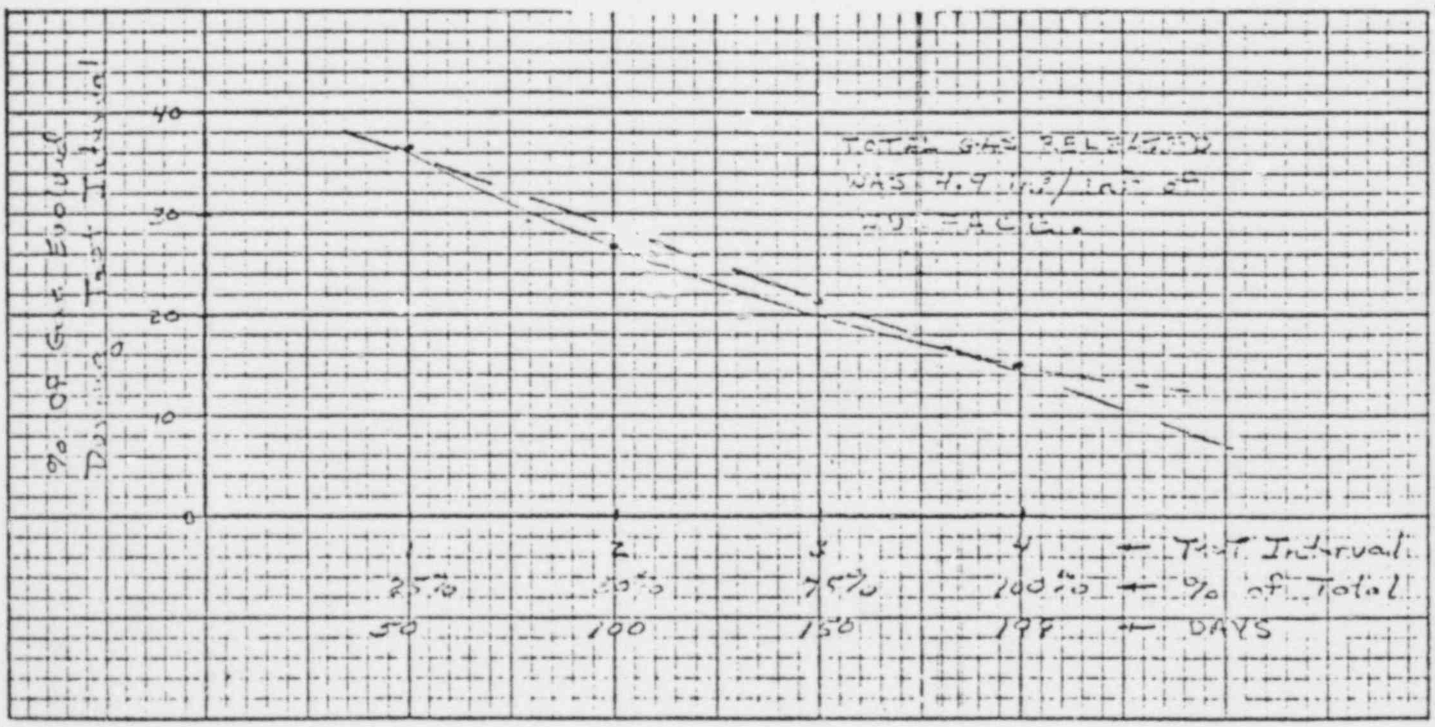
The irradiation of 8.53×10^{17} N/cm^2 produced a weight loss of about 6% with a decrease in all of the dimensions.

4. Prel. Report: Exposure of BISCO Boraflex I to High Temperature Borated Water

Samples containing 50% B_4C were immersed in 240°F borated water (3000 ppm boron) for over 4700 hours (about 200 days). The water pH was adjusted with sodium hydroxide to a range of 9.0 to 9.5.

The tests were interrupted at intervals of 40 days, 80 days, 150 days and 199 days for measurement of the physical parameters. Some of the data is presented on pages 6 and 7.

The data shows that while the sample dimensions decreased by about 1%, the sample mass increased initially by about 0.8% and then decreased but remained greater (by about 0.25%) than the original mass. The density (initially about 114.6 lbs./ft^3) increased by about 4.6% and gas was evolved. The gases were identified as hydrogen, methane, ethane, and carbon dioxide, but the ratio of each was not determined. The gas evolution decreased as a function of time as shown in the following figure:



5. <u>The Effect of Combined Gamma and Neutron Radiation on the Hydrogen Content of BISCO NS-II Neutron Shielding Material</u>

This material is intended to be used to attenuate high energy neutrons escaping from the area between a reactor vessel and the primary shield wall of a pressurized water reactor. This is a silicone resin material having a relatively high hydrogen content.

The tests were conducted at the University of Michigan. Cumulative irradiation was greater than 2×10^{11} rads with the gamma component exceeding 2×10^{10} rads and the integrated fast neutron dose was in excess of 10^{18} N/cm^2 (where e = 1 to 10 Mev).

The following data was obtained:

	Control Sample	Irradiated Sample
Specific Gravity	1.156	1.219
% Hydrogen (weight)	5.69	5.63
Hydrogen density (g/cc)	0.0658	0.0686
Carbon, %	41.21	42.68
Silica, % of sample	61.31	55.50
Oxygen, %	--	25.76

ATTACHMENT 3

Job 1047
2-18-78

EFFECT OF RADIATION ON BISCO NSI

EXPOSED TO A COBALT 60 SOURCE

I. Dose Megarads	Tensile (PSI)	Elongation %	Elastic Modulus
0	510	68	750
16	516	55	938
60	550	40	1375
111	504	38	1326
164	553	23	2404
713	896	3.3	27,151

Sample: .1" x 1" Tensile Bar pulled @ 10 inches/minute

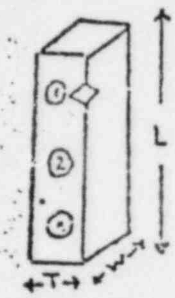
II. Dose Megarads	Stress for 20% Compression (PSI)	(%) Dynamic Comp. Set at 20% Comp.
0	19	2.5%
14	206	0
68	396	0
119	652	0
485	2756 (shattered)	100

Sample: 1.125" Dia x 1" thick button compressed 20% @ 1 inch/min.

BISCO Silicon – 50% B_4C Sample

Long Term Irradiation

Test Sequence		50% B_4C Sample
Pre-Irradiation Weight (gm)		7.0
Pre-Irradiation Dimensions (in)		
	T1	0.271
	T2	0.272
	T3	0.274
	W1	0.317
	W2	0.303
	W3	0.316
	L	3.020



237 Hour Irradiation	
Neutron Dose (N/cm^2)	8.53×10^{17}
Gamma Dose (Rads)	7.11×10^{9}
Gas Evolution	
Cylinder Pressure Buildup	
Rate (PSI/hr)	0.2
Hydrogen (%)	40.89
Oxygen (%)	5.93
Nitrogen (%)	33.72
Methane (%)	19.50

Post Irradiation Weight (gm)		6.6
Post Irradiation Dimensions (in)		
	T1	0.262
	T2	0.262
	T3	0.264
	W1	0.311
	W2	0.301
	W3	0.309
	L	2.950

Attest: [signature] Date: 12/7/76

TABLE II

Dimensional Stability of Boraflex I (% Change from Original) based on Single Long Dimensional Change

SAMPLE#	40 days	80 days	150 days	199 days
1	0.00%	-0.93%	-0.93%	-0.93%
2	0.00	-0.93	-0.93	-0.93
3	0.00	-0.93	-0.93	-0.93
4	0.00	-0.93	-0.93	-0.93
5	0.00	-0.93	-0.93	-0.93
6	0.00	-0.93	-0.93	-0.93
Average	0.00%	-0.78%	-0.93%	-0.93%

TABLE III

Mass Stability of Boraflex I (% Change from Original)

SAMPLE#	40 days	80 days	150 days	199 days
1	+0.82%	+0.83%	+0.59%	+0.47%
2	+0.89	+0.59	+0.41	+0.40
3	+0.81	+1.86	+0.52	+0.35
4	+1.03	+0.75	+0.59	+0.36
5	+0.79	+0.64	+0.48	+0.27
6	+0.07	-0.08	-0.43	-0.44
Average	+0.74%	+0.77%	+0.36%	+0.24%

TABLE IV

Density Stability of Boraflex I (% Change from Original)

SAMPLE#	40 days	80 days	150 days	199 days
1	+0.82%	+1.60%	+1.87%	+5.27%
2	+0.42	+1.11	+1.88	+5.21
3	-0.10	+2.06	+1.93	+4.66
4	+1.44	+1.23	+1.06	+4.19
5	+1.21	+1.59	+1.42	+4.10
6	+0.32	-0.09	+1.07	+4.30
Average	+0.69%	+1.25%	+1.54%	+4.62%

Gas evolution of the Boraflex I samples was continuously monitored as described in the test procedures and reported in the following tables:

TABLE V

Accumulated gas volume evolved (Cubic inches per Sq. In. of sample area)

TIME (days)	TOTAL EVOLVED GAS (in^3/in^2)
40	1.48
50	1.78
80	2.61
100	3.09
150	4.15
199	4.90

Interrogatory 5.1

Should the need arise, what is the shortest practical time needed to remove and ship offsite 1,000 or more spent fuel assemblies?

RESPONSE:

Since it would not be necessary to remove large numbers of fuel assemblies from the plant in a short time period, no such study has been conducted.

Interrogatory 5.2

How many shipping casks will be available in the year 2000?

RESPONSE:

This information is not available.

Interrogatory 6.1

Does Point Beach Nuclear Power Plant monitor the ground water beneath its boundaries for traces of radioactivity?

RESPONSE:

The onsite well located just south of the switchyard is sampled quarterly for laboratory analysis. In addition, lakewater is sampled on a monthly basis at five points along the shoreline, the natural terminus of groundwater gradients in the area.

Interrogatory 6.2

Is it possible that spent fuel pool coolant water could leak through the liner and concrete base without being collected in the leakage detection bucket?

RESPONSE:

It is possible that leakage through the liner would not reach the leakage collection tank. A small leak would be totally consumed in wetting the concrete surface of the leak channels and by evaporation en route to the collection tank.

Leakage could likewise bypass the collection channels if there were open cracks in the concrete under the liner. Due to the four to five foot thick reinforced concrete construction of the pool bottom and walls, it is not considered likely that cracks large enough to readily transmit pool water can develop. No significant open cracks have been observed on the outside surface of the pool walls.

Also, leakage in the fuel transfer canal may flow either to where the channels are drained to the collection tank or out onto the ground floor of the facades if the leak is in the sloping end walls of the canal. The facades are controlled areas and drainage there is collected and monitored before disposal.

The Point Beach spent fuel pool is designed and constructed to facilitate detection of significant leaks that might develop over the life of the plant. The leak collection system is a feature that is not present in all plant designs and in our opinion provides an effective system for continuous verification of pool liner integrity.

Interrogatory 6.3

Has the collection bucket ever collected pool water leakage?

RESPONSE:

There was a leak in the fuel transfer canal liner in 1974. Water was collected in the fuel transfer canal collection system and drained into the collection tank. Seepage was also present on the Unit 2 facade floor. The leak was stopped by dewatering the canal and repair welding a defective liner weld. The leaking weld was on the liner penetration of the fuel transfer tube butterfly valve operating rod.

Interrogatory 6.4

Has the leakage detection system ever been tested? How is it tested?

RESPONSE:

The leakage detection system has never been tested, except by its performance in-service. The system consists of a grid of open channels in the concrete under the liner on the floor and walls of the pool and transfer canal. The channel grids form separate systems in the north pool, the south pool, and the transfer canal. Each grid system is piped to an accessible location in the Auxiliary Building where the water collected in the channels can drain into an open collection tank. The only meaningful test of the system would be to drill holes in the completed liner at numerous locations and verify that leakage was collected in the collection tank. A large number of holes would be required to verify that the system performs in all areas of the pool. Because this is a simple gravity system that required only that the water run to the low point and drain into the collection tank, initial testing was not required. There are redundant flow channels to the point where the water is drained to the collection tank, so plugging of a single channel will not prevent water from getting to the collection tank.

Initial testing was performed on the 3/16" thick stainless steel liner plate. All accessible welds were tested with a vacuum box. Welds that were inaccessible for vacuum box testing were subject to liquid penetrant examination in accordance with ASME Section III. All leaks, regardless of size, were repaired. Prior to acceptance of the liner, the pool and canal were filled with water to verify leak tightness through the monitoring channels.

Interrogatory 7.1

Contract(s) and/or correspondence pertaining to offsite placement and/or disposal of spent fuel racks, filters, and other increased quantities of low-level radioactive wastes due to the compaction of spent nuclear fuel as proposed in the request for a license amendment. Indicate where these increased quantities of low-level radioactive solid wastes are to be placed.

RESPONSE:

As indicated in the answer to 3g.1, a final decision regarding the disposal of spent fuel racks has not been made; hence, there are neither contracts nor correspondence pertaining thereto. For a discussion of postulated increased wastes, please refer to the answer to NRC question B-6 in our July 19, 1978, submittal. The negligible additional quantities, if any, will not require special contracts or correspondence and would be shipped to one of the currently available burial sites: Barnwell, S. Carolina; Richland, Washington; or Beatty, Nevada.

Interrogatory 7.2

Study(ies) and/or correspondence regarding the feasibility of land burial and/or storage (engineered or otherwise) of the increased amounts of low-level radioactive solid wastes due to compaction of spent fuel at the Point Beach site.

RESPONSE:

Please refer to the response to NRC question B-6. The negligible quantities involved require no special studies or correspondence. There are no plans for on-site disposal or indefinite storage of low-level wastes at Point Beach Nuclear Plant.

Interrogatory 8.1

Study(ies) or analysis(es) delineating the examination and condition of each Point Beach spent fuel assembly presently in storage at Nuclear Fuel Services in West Valley, New York and slated for removal to the Point Beach Plant.

RESPONSE:

Prior to loading any spent fuel for shipment to Point Beach, Nuclear Fuel Services will perform a visual inspection; no destructive examination of spent fuel assemblies is contemplated.

During 1978, six of the original 120 Point Beach fuel assemblies in storage at West Valley were shipped to Battelle Northwest Laboratories (BNWL) in Hanford, Washington, for use in a testing program sponsored by the U. S. Department of Energy. These assemblies were selected by BNWL without prior physical examination and included fuel which had been in storage since October 1972. Shipments were made in an NAC-1 shipping cask; should return of spent fuel from NFS ultimately be required, shipments to Point Beach would be made using a cask of the same design. All assemblies received by BNWL from NFS were considered safe and intact and exhibited no signs of physical deterioration.

Interrogatory 11.1

What are the projections of population growth through the year 2000 upon which you have based your data determining the safety of radiation levels for non-occupational exposures?

RESPONSE:

Population projections through 1985 were provided in Section 2.4 of the Point Beach Final Facility Design and Safety Analysis Report and no further projections have been made.

Interrogatory 11.2

What radiation exposure monitoring will be done on people who live in the vicinity of the plant in light of the increased and longer term storage of spent fuel?

RESPONSE:

Monitoring on people who live in the vicinity of the plant is neither planned nor required. Current monitoring of all effluent streams and potential dose pathways in the environment will continue. As specified in Section 8.0 of the Application, in the responses to NRC questions B-6 and B-7b, and in the response to Interrogatory 1.4, releases due to the increased storage of spent fuel are not expected to be significant.

Interrogatory 16a.1

Detailed analyses of the effects of borated water on Point Beach spent fuel, cladding, support frames, storage racks, fuel basin liner, neutron absorber plates and all other components in contact with the storage pool borated water for the period of license. Please supply this information.

RESPONSE:

Detailed analyses of the effects of the borated water on components in the spent fuel pool have not been performed. The materials in contact with the borated water (which is a weak acid) are generally known to be corrosion resistance. Reference 1 identifies the stainless steels and other materials as having a good chemical resistance to corrosion when exposed to boric acid.

It should be of interest to note that mild solutions of boric acid (around 1000 ppm; which is about 1/2 of the spent fuel pool water concentration) is commonly used in households as eyewashes or wound cleansing solutions. Obviously, it is not a highly corrosive chemical.

The materials that are being placed into the spent fuel pool due to this modification are austenitic stainless steel and the Boraflex poison material. These materials are compatible with the spent fuel pool environment.

The Point Beach fuel and cladding is designed for use in a borated water environment in the operating reactor under conditions very much more severe than that which will be experienced in the spent fuel pool. In the operating reactor the fuel and cladding is designed to be exposed to neutron irradiation, temperatures above 600°F, pressures of 2250 psia without significant corrosion or loss of fuel rod cladding integrity. In the relatively mild spent fuel pool environment any deleterious effects of borated water on fuel and cladding is reduced to relative insignificance even if the spent fuel pool temperature should increase substantially.

Ref. 1 "Chemical Engineers' Handbook", Third Edition,
pages 1461 and 1473.

Interrogatory 16a.2

Will an increased spent fuel pool coolant water temperature cause an increase in the corrosion rate on pool components?

RESPONSE:

It is anticipated that the pool water temperature will not exceed 120°F for any appreciable length of time during the plant's lifetime. With respect to corrosion, 120°F is a low temperature and corrosion effects are not considered to be significant. In general, an increase in temperature will result in an increase in corrosion. However, an increase in temperature also substantially decreases the solubility of oxygen in water(1), and generally the rate of corrosion is roughly limited by the rate of diffusion of dissolved oxygen to the metal surface(2).

The corrosion resistance of austenitic stainless steels (Type 304 is in this category) has been researched extensively, and the low corrosion rate of austenitic stainless steels is one of the principal reasons why these steels are used extensively in nuclear power plants.

(1) Marks, "Standard Handbook for Mechanical Engineers", page 6-126.
(2) Ibid, page 6-125.

Interrogatory 16b.1

Analysis of the effects of accelerated corrosion, microstructural changes, alterations in mechanical properties, stress corrosion, cracking, intergranular corrosion, and hydrogen absorption and precipitation by aluminum alloys due to compaction of spent fuel as proposed at Point Beach for the duration of the operating license.

RESPONSE:

The Applicant has not proposed installing any aluminum alloys in the spent fuel pool. There are no items constructed of aluminum alloys permanently installed in the spent fuel pool.

Interrogatory 16c.2

Describe the metallurgical composition of the storage pool liner, pipes, storage racks and rack bases and provide an analysis of the electrolytic corrosion effects of the dissimilar alloys on the spent fuel, its cladding and all components of and in the spent fuel storage pool.

RESPONSE:

The spent fuel pool liner plate was fabricated from hot-rolled and annealed ASTM A-167, Type 304L stainless steel material.

The spent fuel pool cooling piping was fabricated from ASME SA-312 Type 304 stainless steel material.

The spent fuel storage racks are to be fabricated from cold-rolled ASTM A-240, Type 304 stainless steel material.

The storage rack base materials will be Type 304 stainless steel materials probably meeting the requirements of ASTM A-240 for plate material, A-479 for bar material, and either A-312, A-358, or A-409 for pipe materials.

The above referenced standards are material standards generally available to the public and are industry recognized standards. The standards present requirements for material chemistries (maximums, minimums, and/or ranges), material properties (minimum tensile strength, etc.), testing requirements, as well as other requirements.

All of the identified materials are basically 304 stainless steel (304L has a lower maximum carbon content, may have more nickel in it, and is not quite as strong; the 0.2% offset yield strengths at 300°F are about 25,800 psi for 304L and 26,400 psi for 304). Thus, they are not dissimilar materials and there should be no general electrolytic corrosion.

Interrogatory 16d.3

Describe the proposed monitoring of the more densely stored and increased amounts of spent fuel assemblies at Point Beach and provide an analysis of the feasibility of monitoring individual spent fuel assemblies to detect defects and leakage of spent fuel due to loss of integrity of the cladding.

RESPONSE:

The spent fuel pool water is monitored on a regular basis by laboratory analysis of water samples. Beyond that, it is neither necessary nor desirable to monitor individual spent fuel assemblies. The integrity of the cladding of the fuel assemblies is expected to remain in the same condition as when they were removed from the reactor. No cladding failures due to long term storage are expected. Leaks in spent fuel assemblies would be insignificant because the isotopes of xenon and iodine have decayed away within one year after discharge, and there is no longer a temperature differential sufficient to provide a mechanism to drive nuclides out of the fuel rods. Moreover, an individual fuel assembly monitoring program would require that the fuel be moved to the monitoring device. This would result in a considerable increase in spent fuel movements with no benefits derived from the movements.

Interrogatory 16e.4

Delineate provisions made for encapsulating defective spent fuel assemblies in the event of defect, leakage and/or loss of integrity.

RESPONSE:

The encapsulation of defective spent fuel assemblies in the event of defect, leakage and/or loss of integrity due to increased storage is unnecessary. See the response to Interrogatory 16d.3.

Interrogatory 16f.5

Studies which quantify the anticipated thickness of crud layers on the spent fuel assemblies and evaluates the tendency of the crud layers to influence spent fuel cladding corrosion resulting from the more densely stored and increased quantities of spent fuel at Point Beach.

RESPONSE:

We do not have any studies which quantify the anticipated thickness of crud layers on the spent fuel assemblies and evaluates the tendency of the crud layer to influence spent fuel cladding corrosion resulting from the more densely stored and increased quantities of spent fuel at Point Beach.

Interrogatory 16g.6

Analysis of the state of art and availability of other more densely stored spent fuel assemblies in water storage pools at Point Beach, including on-site and off-site dry storage in sealed storage casks, air cooled vaults, near surface heat sinks. Evaluate the health, safety, environmental and economic impact of these alternatives and compare with the impacts of more densely-racked underwater storage at Point Beach.

RESPONSE:

Applicants have considered the economic and technical feasibility of alternatives. None of the alternatives provide either economic advantage or advantages in maintaining the integrity of the spent fuel.

In the early stages of its evaluation of spent fuel storage alternatives, Wisconsin Electric discussed with Stone and Webster Engineering Corporation the feasibility of constructing a separate wet-storage facility, interconnected with existing plant systems, on the Point Beach site. This type of facility was discussed in a Stone and Webster topical report which was accepted for review by the U. S. Nuclear Regulatory Commission in January 1977. The capacity of the proposed facility was to have been 1700 assemblies and the total cost (not including storage racks) was estimated to be $23,305,000. On the basis of this cost estimate and the judgment that there were no tangible benefits associated with storage of spent fuel in a separate facility, the concept of a separate storage pool at the reactor site was rejected in favor of modification to the existing storage pool.

The alternative of dry-storage for Point Beach spent fuel was not recommended for several reasons. The technology for dry-storage of spent fuel discharged from commercial PWR's (pressurized-water reactors) has not been developed to the point where it could have been expected to provide relief during the time frame necessary for continued operation of the Point Beach Nuclear Plant. Dry-storage concepts have been proposed in the past as potential solutions to the permanent disposal issue, but the interim storage of spent fuel in a "dry" facility is an idea which has recently begun to receive more attention.

A study involving the dry-storage concept was recently conducted by Allied Chemical Corporation and Ralph M. Parsons Company for the Idaho Chemical Processing Plant (ICPP). A cost/benefit analysis was made of the following alternatives:

1. A conventional storage facility employing underwater handling and storage
2. An air-cooled storage facility employing remote fuel handling and storage
3. A water-cooled storage facility employing remote fuel handling and storage

The alternatives were evaluated on the basis of operating/maintenance considerations, safety issues, and cost/schedule expectations. An architect/engineer's comparison of cost/schedule expectations for each of the alternatives (from the point of final facility design) showed no significant differences in the costs or schedules. On the basis of the remaining factors, the conventional deep-pool concept (alternative 1) was chosen for the ICPP. The air-cooled storage concept (alternative 2) was the least-preferred of the alternatives evaluated. Since the dry-storage concept is not preferred in this study over wet-storage on the basis of a technical evaluation, and since the cost/schedule expectations are comparable to those for a separate wet-storage facility (either on-site or AFR), there does not appear to be any basis for resorting to construction of a separate dry-storage facility.

Interrogatory 16h.7

Analysis of anticipated problems in handling spent fuel during the period of license due to loss of integrity of the spent fuel and its cladding in more densely-stored and increased quantities of spent fuel.

RESPONSE:

No problems are epxected in handling spent fuel during the period of license due to loss of integrity of the spent fuel and its cladding in more densely-stored and increased quantities of spent fuel. The integrity of the cladding and structural capability of the fuel assemblies are expected to remain in the same condition as when they were removed from the reactor. No detrimental effects due to long term storage are expected. Therefore, no analysis of handling problems has been performed.

Interrogatory 16i.8

Delineate the chemical makeup of spent fuel storage pool components at Nuclear Fuel Services in West Valley, New York. Compare this information with data on the chemical makeup of spent fuel storage pool components at Point Beach. If there are chemical differences in the Point Beach and NFS storage pool components, indicate the anticipated corrosive effects of these chemical differences on the spent fuel, its cladding, and other components of and in the Point Beach spent fuel storage pool due to the transfer and placement of the Point Beach spent fuel now in storage at NFS in the Point Beach storage pool.

RESPONSE:

Water in the storage pool at Nuclear Fuel Services (NFS) in West Valley, New York, is characterized as demineralized water. This characteristic is maintained by running pool water through demineralizers utilizing ion exchange resins (which are periodically replaced). Water temperature in the storage pool is maintained between 70°F and 79°F. The Point Beach spent fuel pool contains borated demineralized water.

Point Beach fuel assemblies have been exposed to both demineralized water and borated water while residing in the core (and at extremely high temperatures) without experiencing significant corrosive effects. The temperature of the storage environment and the heat generation of the assemblies, themselves, are only a fraction of what they would be in the core of a nuclear reactor. No adverse corrosive effects are anticipated as a result of differences in the chemical makeup of spent fuel storage pool components at Point Beach and NFS; therefore, no specific analysis has been made.

Roger A. Newton

Roger A. Newton
Senior Nuclear Engineer

Subscribed and sworn to before me
this 27th day of October, 1978

Notary Public, State of Wisconsin

My commission expires July 6, 1980

October 27, 1978

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of WISCONSIN ELECTRIC POWER COMPANY (Point Beach Nuclear Plant, Units 1 and 2)	)))))))	Docket Nos. 50-266 50-301 Amendment to License Nos. DPR-24 and DPR-27 (Increase Spent Fuel Storage Capacity)

AFFIDAVIT OF SERVICE

I hereby affirm that copies of "Applicant's Answers to Interrogatories Propounded by Intervenor on September 27, 1978", were served upon those persons on the attached Service List by deposit in the United States mail, postage prepaid, this 27th day of October, 1978.

Roger A. Newton

Roger A. Newton
Senior Nuclear Engineer

Subscribed and sworn to
before me this 27th day
of October, 1978

Dorothy M. Eichmann

Notary Public
State of Wisconsin

My commission expires

July 6, 1980.

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