50-266
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November 1, 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 G 50-301 Amendment to License Nos. DPR-24 and DPR-27 (Increase Spent Fuel Storage Capacity)

APPLICANT'S ANSWERS TO INTERROGATORIES
PROPOUNDED BY THE STATE OF WISCONSIN
ON OCTOBER 2, 1978

Interrogatory 1

Please state the type of airborne radioactive emissions expected from the spent fuel pool. How will these emissions increase in quality and quantity as a result of the increased fuel expansion? What model is utilized in making the quantative calculation as to the effect of the interim expansion?

RESPONSE:

The potential for airborne radioactive effluents from stored fuel is discussed in Section 8 of Attachment A to the Application. As discussed therein, the only two isotopes which have any potential for increasing with increased storage 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. Dose calculations are consistent with NRC Regulatory Guide 1.109 and meteorological calculations are consistent with NRC Regulatory Guide 1.111.

(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 released 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 assuming 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 releases of airborne effluents from the pool. Thus, the incremental number of curies calculated above would not be expected to actually be released.

Interrogatory 2

Do you plan to increase the air monitoring capability inside of the pool containment structure? If not, why not? If your answer is yes, please describe the contemplated increase.

RESPONSE:

The present air monitoring capabilities are appropriate for both present and planned storage. An increase in the capabilities is neither planned nor needed. See response to Interrogatory 1.

Interrogatory 3

What is the calculated radioactive dose rate to a person standing next to the spent fuel storage pool after the expansion? What model is utilized in calculating this dose?

RESPONSE:

The maximum dose received by a person standing at the edge of the spent fuel pool is virtually the same as the doses stated for the surface of the pool in Section 7.4 of Attachment A of the Application. The doses were calculated by a point kernel technique used in a modified QAD computer program. QAD is the generic designation for a series of point-kernel computer programs designated for calculating the effects of gamma rays that originate in a volume-distributed source. The QAD approach has been accepted nationally and is used with modifications appropriate to local computer hardware and specific applications.

Interrogatory 4

What is the probability that the radioactive releases from the Point Beach Nuclear Power Plant will combine with those from the Kewaunee Nuclear Power Plant? What meteorological conditions would have to exist for the radioactive plumes from these two plants to come in contact with each other and intermingle? Please set forth the model which you base your estimate upon.

RESPONSE:

It is meteorologically impossible for the effluent plumes from both plants to intermingle between the plants, since the wind cannot be blowing north and south at the same time. In the event of a south wind, it is possible for some combination of effluents to occur at some point north of both plants. At such a point, the contribution from the southernmost plant would be negligible because of the diffusion achieved over the rather substantial distance involved. A similar observation can be made for the converse case of a north wind and some point south of both plants. However, the more conservative case occurs at some point between the two plants, not from simultaneous contributions of both plants but from the alternate contributions from either plant dependent upon the meteorological frequency involved. To demonstrate that the cumulative effects of increased storage of spent fuel are negligible, we have considered 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, 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.

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 5

What would be the maximum water temperature reached within the spent fuel pool were cores from both nuclear facilities at Point Beach offloaded into the pool? In calculating this temperature, please assume that the offload of both cores would result in the spent fuel pool being full. Please state the model utilized in calculating this temperature, as well as any assumptions relied upon. How would an increase in the water temperature in the spent fuel pool affect the quantity and quality of radioactive emissions from the pool? Were a temperature increase in the pool to result in boiling and a loss of coolant, how would the quality and quantity of radioactive emissions from the pool be affected? Please state the model utilized and assumptions relied upon in making this determination. What would be the radiation dose received by a person standing next to the spent fuel pool during such a rise in temperature?

RESPONSE:

Under the postulated conditions, 1260 storage positions would be filled with 242 spaces available for unloading the two cores. The 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 13 days are required to unload a core and that the cores are unloaded sequentially with one day pause between unloadings:

a. decay heat from the second core unload 13 days after shutdown of reactor (from Figure 1) — 11.5×10^6 BTUs/hr

b. remaining decay heat from the first core unload 27 days after shutdown of reactor (13 days for unloading the first reactor, 1 day pause, 13 days for unloading 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 this 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.

Since the pool water temperature is not expected to exceed the normal temperature of 120°F, there would be no affect on the quantity and quality of radioactive emissions from the pool for the situation assumed in this interrogatory.

If one were to arbitrarily assume the pool water reached the boiling point, 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 designated 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. By comparing this release of tritium with the releases and doses calculated in the response to Interrogatory 1, it is concluded that the maximum dose to an individual living at the site boundary would be insignificant. Administrative procedures and ordinary good health practice would preclude the possibility of a worker continuing to stand at the edge of the pool while it boiled; hence, any significant dose to workers is unlikely.

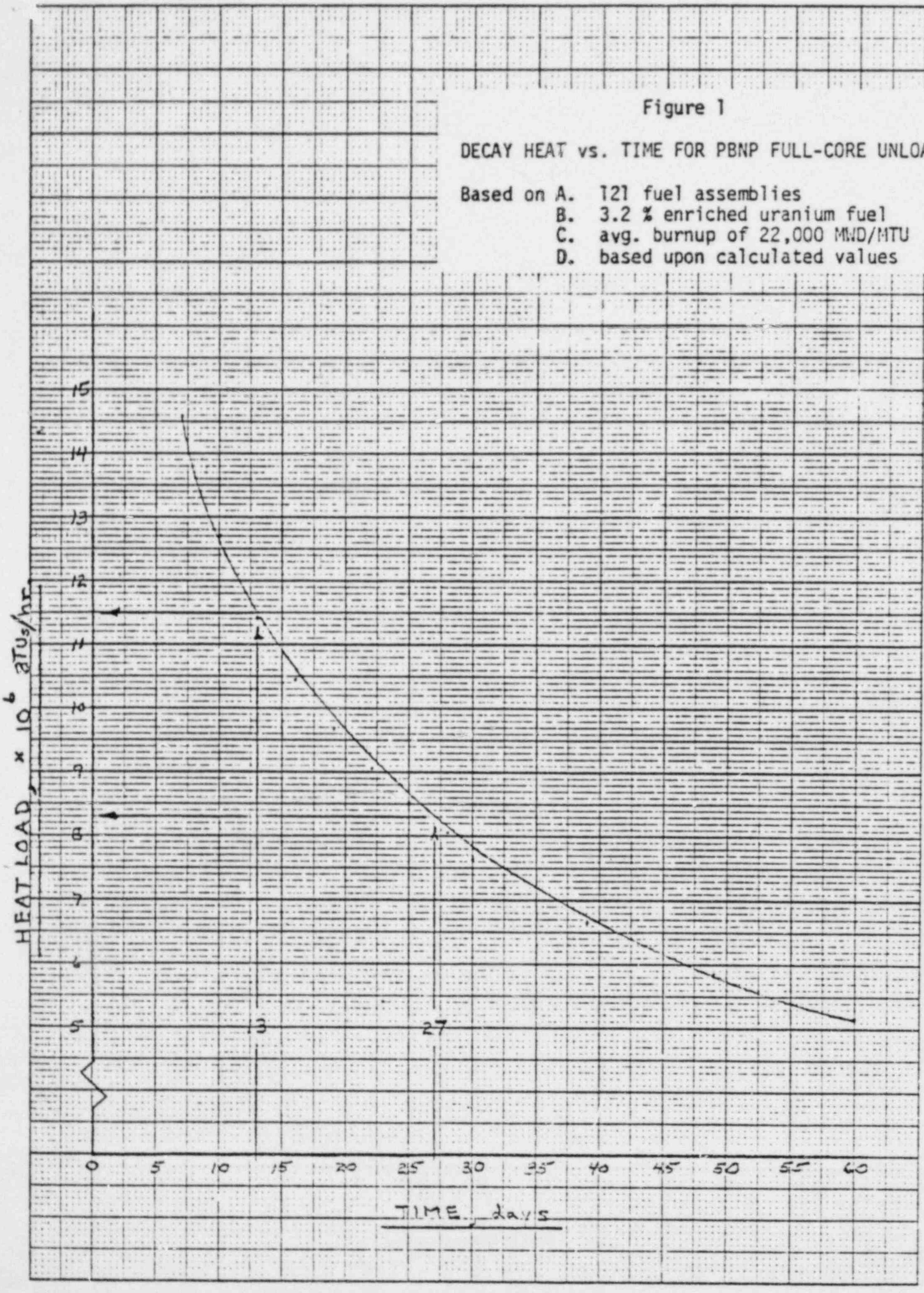


Figure 1

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

Interrogatory 6

What precautions have you taken to prevent the blockage of the coolant inflow and outflow pipes in the spent fuel pools? What would be the effect on the temperature of the pool of a blockage of either the inflow or outflow pipe? Would such a blockage cause an increase in radiaoctive emissions from the pool, due either to the inability to filter the pool's water during blockage or due to increased temperatures during blockage? If not, why not? If your answer is yes, please state the expected increase. Also state the model utilized and the assumptions relied upon in making your determination.

RESPONSE:

The spent fuel pool coolant suction (outflow) 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.

The spent fuel pool coolant discharge (inflow) piping is located in the southwest corner of the south half of the spent fuel pool. The piping enters the pool water vertically, from above, and is terminated ten feet below the normal water surface elevation.

The pool suction and discharge piping are designed so that blockage will not occur. 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 suction pipe opening, it is inconceivable that anything could get sucked down and then up into the pool suction pipe. With respect to the discharge piping, the flow is from the pipe into the pool, and therefore the discharge piping would not become blocked.

The pool suction and discharge piping consists of 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 an eight-inch suction pipe was blocked, the cooling could simply be transferred to the other cooling train. If somehow the ten-inch suction pipe was blocked, the cooling system would 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.

If the cooling system were turned off, the pool water temperature would increase during the time period for cleaning out the blocked line. Figure 2 shows the time required to heat up the spent fuel pool as a function of heat load.

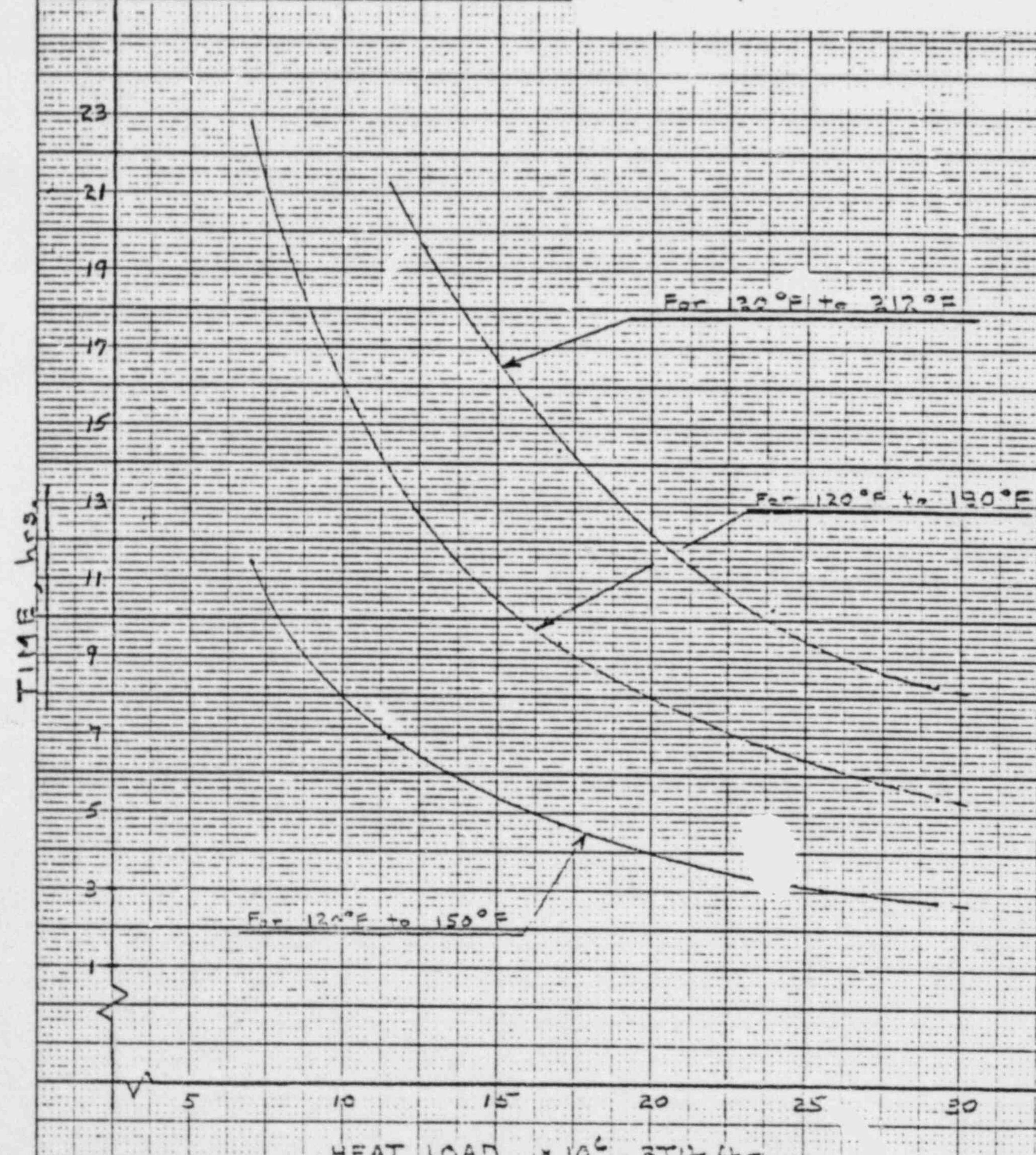
There would be no increase in radioactive emissions from the pool unless the water temperature were to reach the boiling point. Refer to the answer to Interrogatory 5 for the consequences of boiling.

Since filtering is done intermittently, rather than continuously, the inability to filter during the postulated blockage would not effect emissions from the pool.

FIGURE 2

TIME TO HEATUP THE
PBNP SPENT FUEL POOL

(only considers the heatup of water)



Interrogatory 7

If the coolant in the spent fuel pool were to boil away, what would be the radiation dose calculated to a person standing at the pool's edge? Please state the model utilized and the assumptions relied upon in making this calculation.

RESPONSE:

The radiation dose under the conditions specified in this interrogatory has not been calculated. The large loss of pool water and the uncovering of spent fuel is an unacceptable condition which is precluded by the design features of the spent fuel pool and plant. 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 and as a radiation shield. 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.

For boiling to occur in the spent fuel pool, both spent fuel pool cooling trains would have to be inoperable for a period of time. 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 8

What precautions have been taken by you to prevent the possibility of a simultaneous loss of both storage pool coolant pumps? Were both pumps to fail, how long would it take for the coolant in the spent fuel pool to boil? Were both pumps to fail, what emergency measures would you take to prevent coolant boiling from occurring?

RESPONSE:

In the response to Interrogatory 7, the precautions taken in the design of the cooling system to prevent the simultaneous loss of both cooling trains are described As stated in that response, the simultaneous failure of both cooling trains is not considered credible. If one were to arbitrarily assume the failure of both pumps, the heatup times have been calculated and are presented in Figure 2 of the rasponse to Interrogatory 6. Figure 2 shows the time available to restore at least one train to operation before certain temperatures are reached, depending upon the heat load generated by the spent fuel in the pool at the time of postulated simultaneous pump failures. Obviously, the plant Maintenance Department (mechanical or electrical) would be required to restore the operating capability of at least one train as soon as possible commensurate with the situation. Spare pump parts are maintained on-site and replacement of these parts can be accomplished in a short time period.

Interrogatory 9

How would a loss of coolant in the pool, due to either a fracture of the pool liner or boiling away of the coolant, affect the integrity of the spent fuel storage racks, due to increased thermal stresses? Please assume in your calculation that the racks are filled with fresh spent fuel directly from the core. Please state the model utilized and other assumptions relied upon in making these calculations.

RESPONSE:

A large loss of pool water and the uncovering of fuel is precluded by the design features of the pool and the plant. See response to Interrogatory 7.

In 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 - 1,000 gpm indefinitely.

Interrogatory 10

What is the probability that a fuel assembly dropped during loading would crack or otherwise damage the pool liner?

RESPONSE:

While it is possible, but extremely unlikely, that a dropped fuel assembly could damage the pool liner, no calculation of the probability of such an occurrence has been made. It should be noted, however, that the probability of liner damage will be substantially decreased by the reracking program because the area of the pool liner potentially exposed to a dropped fuel assembly will be decreased.

Interrogatory 11

In the event of damage to the pool and/or pool liner while the spent fuel pool is filled to capacity, how would repairs be made? Would repair necessitate removing the stored fuel assemblies from the pool? If so, where would these fuel assemblies be kept during repair?

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 12

In your response to Question A-12, you cite long-term radiation studies documented in BISCO Report 1047-1. Under what conditions, were these studies conducted; by "conditions", I am referring to the gamma flux to which the boraflex I plates were subjected, the time period in which they were subjected to gamma flux, and the medium (water, air, etc.) in which the experiments took place.

RESPONSE:

A summary of the results of previous testing of 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.

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. $2500 \frac{in^2}{slab} \times 1500 \text{ slabs} \times 5.0 \frac{in^3 \text{ of gas}}{in^2 \text{ of slab area}} = 18.750 \times 10^3\ in^3$ of gas

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

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

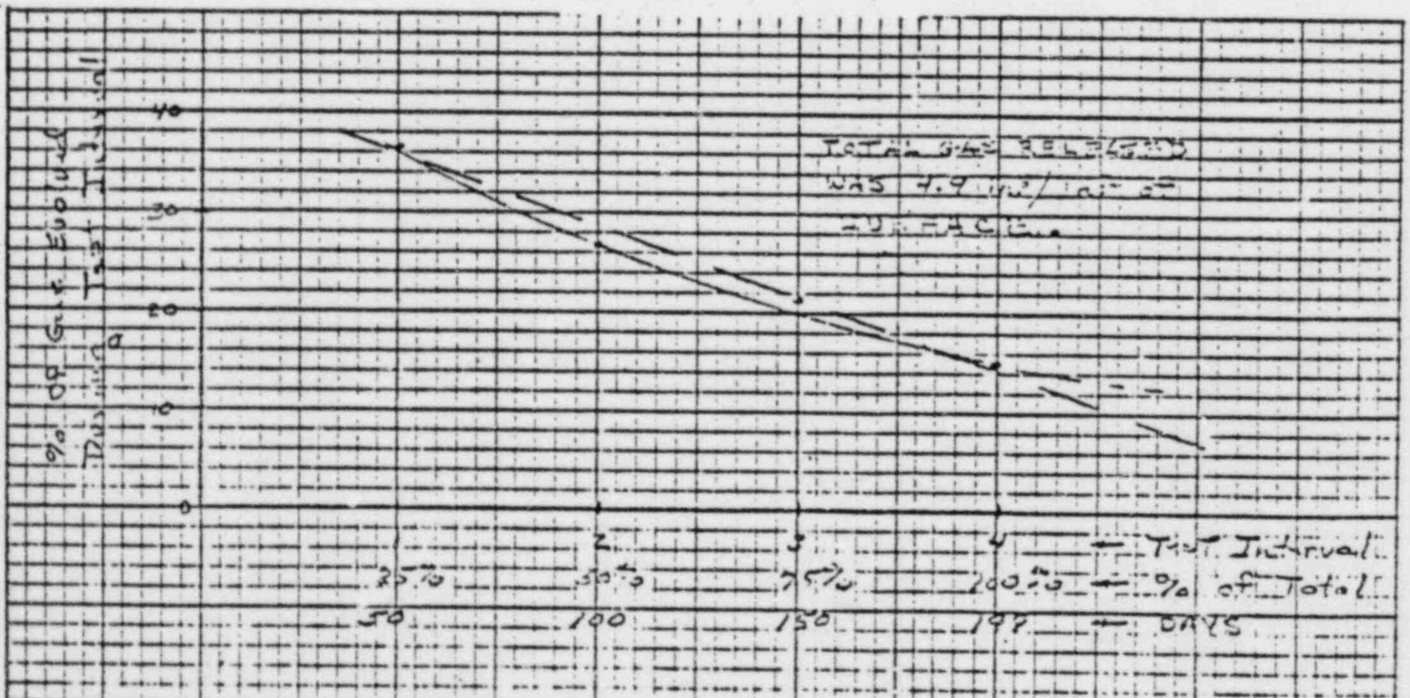
The irradiation of 8.53×10^{17} N/cm^2 produced a weight loss of about 6% with a decrease in all of the dimensions.

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
486	2756 (shattered)	100

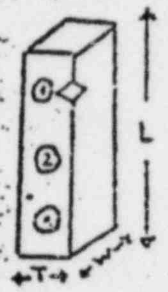
Sample: 1.125" Dia x 1" thick button compressed 20% @ 1 inch/min.

Attachment A
Page 5

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: Reed P. Bunn   Date: 12/7/76

TABLE II

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

SAMPLES	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)

SAMPLES	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 13

Do you presently monitor the groundwater around the Point Beach Nuclear Power Plant for radioactivity? If your answer is no, do you plan to install groundwater monitoring equipment to monitor releases from the spent fuel storage pool after expansion? If not, why not?

RESPONSE:

Groundwater beneath the site is sampled on a quarterly basis at the plant well, located just south of the switchyard. In addition, lakewater is sampled on a monthly basis at 5 points along the shoreline, the natural terminus of groundwater gradients in the area.

Interrogatory 14

If your answer to the above interrogatory is yes, please state whether you plan to increase or change your present groundwater monitoring system in any way, describing the changes contemplated. If you do not contemplate changing your present groundwater monitoring system, state the reasons for this decision.

RESPONSE:

Changes are neither contemplated nor needed. A significant leak would be detected by the presently available indicators: wellwater sampling, shoreline lakewater sampling, pool leak detection system, and indications of unusually high quantities of makeup water to the pool. Further, as explained in Section 7.4 of Attachment A to the Application, increased fuel storage is not expected to result in any significant increase in the radionuclide concentrations in the pool water.

Interrogatory 15

How many fuel assemblies are presently stored at the NFS plant in West Valley, New York? What precautions are planned in order to insure that any fuel assemblies returned from NFS arrive safe and intact? What procedures are planned should a number of the fuel assemblies returned arrive in deteriorated condition?

RESPONSE:

Wisconsin Electric currently has 114 fuel assemblies in storage at the NFS plant in West Valley, New York.

Precautions to insure that spent fuel returned from NFS is safe and intact are addressed in our previous response to NRC Question C-19 in our October 10, 1978, submittal.

Special procedures to specifically deal with fuel assemblies which are returned and arrive in a deteriorated condition are not required. The fuel assemblies are intact at NFS and are expected to show no signs of physical deterioration resulting from storage and shipment. This is borne out by the general experience of the nuclear industry in respect to the integrity of spent fuel while in water pool storage and during transport, including our recent experience of shipping six fuel assemblies from NFS to Battelle Northwest Laboratories (BNWL) in Hanford, Washington. In any event, even if a returned fuel assembly did arrive in a deteriorated condition, the potential releases and contamination to pool water would be far less than the increased contamination to pool water that normally occurs at each refueling. The spent fuel pool filter and cleanup system would remove any contaminants from the pool water in a short time period.

Interrogatory 16

What is the expected increase in occupational exposure due to the daily operation of the expanded spent fuel disposal pool? Please state the assumptions relied upon in making this calculation.

RESPONSE:

As discussed in Section 7.4 of Attachment A to the Application, radiation dose rates with the expanded storage capacity are not expected to be significantly different from those encountered for the present design. In fact, for one particular case discussed in Section 7.4, the dose rate will actually go down. Furthermore, no additional worker activity in the vicinity of the spent fuel pool is anticipated as a result of the increased storage capacity. Hence, there will be no increase in occupational exposure due to the daily operation of the expanded spent fuel pool.

Interrogatory 17

What is the expected increase in radiation exposure to the public due to operation of the expanded spent fuel storage pool? Please state the assumptions relied upon in making this calculation.

RESPONSE:

See the response to Interrogatory 1.

Interrogatory 18

State the technical basis upon which you believe that the spent fuel stored in the pool will retain its integrity for the entire period of licensing.

RESPONSE:

The Point Beach fuel and cladding are designed for use in a borated water environment in the operating reactor under conditions much more severe than that which will be experienced in the spent fuel pool. In the operating reactor the fuel is designed to be exposed to neutron irradiation, temperatures above 600°F, and pressure of 2250 psia without significant corrosion or loss of fuel rod cladding integrity. For in-reactor corrosion rates at temperatures of 500°F, it would take approximately 2,200 years to penetrate the Zircaloy-4 cladding. Since all of the structural material of the fuel assembly have the same or better corrosion resistance characteristics than Zircaloy-4, the structural integrity of the assembly would also remain intact for as long as that of the fuel cladding.

In the relatively mild spent fuel pool environment, any deleterious effects of the borated water on fuel and cladding are reduced to relative insignificance even if the spent fuel pool temperature should increase substantially.

The good performance of fuel stored underwater in pools is supported by extensive successful experience with storage of spent fuel in water pools as discussed in the Draft Environmental Impact Statement prepared by the U. S. Department of Energy; DOE/EIS-0015-D, Storage of U. S. Spent Power Reactor Fuel, August 1978, wherein it states: "The technology of water-cooled basin storage is well developed, and water basins have been successfully used for receiving and storing spent nuclear fuel since the beginning of the nuclear age, more than

30 years ago. Spent fuel has been stored without any significant incident or detriment to the surrounding environment or population. Further, the storage has been accomplished without any serious deterioration of the fuel cladding.(1)"

(1) A. B. Johnson, Behavior of Spent Nuclear Fuel in Water Pool Storage. USERDA Report BNWL-2256, Battelle Pacific Northwest Laboratories, Richland, Washington (September 1977)

Interrogatory 19

Please state the average, median and maximum burnup of the spent fuel which will be stored in the fuel pool. How does the burnup of the fuel affect your estimate of long-term fuel integrity? Please be specific. Please state the name of all technical studies and/or experiements with which you are familiar, whether completed or ongoing, which assess the integrity over a forty-year period of spent fuel having a burnup as high as that of the spent fuel with the maximum burnup expected to be placed within the Point Beach spent fuel pool.

RESPONSE:

It is not possible to state the precise average, median and maximum burnup of spent fuel which will be stored in the spent fuel pool, because future fuel assembly and fuel cycle designs have not yet been specifically developed. Typically, discharged fuel has achieved burnups ranging from about 21,000 MWD/MTUo up to 40,000 MWD/MTUo. Currently, the region average discharge burnups are targeted on a burnup of 33,000 MWD/MTUo with expected maximum and minimum burnups of about 37,000 MWD/MTUo and 28,000 MWD/MTUo respectively. Higher region average discharge burnups may be achieved in the future. Higher burnups would have little, if any, incremental impact on spent fuel integrity because the storage duty of the fuel in a spent fuel pool environment is so much less limiting than that experienced in an operating reactor. Fuel temperatures, dynamic loads and thermally induced stresses on all portions of the fuel assemblies will be much lower. Thus, fuel assembly burnup does not have a significant effect on long-term fuel integrity under spent fuel storage conditions. We are not aware of whether or not studies of long-term fuel integrity have specifically included spent fuel having burnups in excess of the maximum expected at Point Beach.

Interrogatory 20

How will the integrity of the fuel rods in the spent fuel pool be monitored?

RESPONSE:

The spent fuel pool water is, and will be, monitored on a regular basis by laboratory analysis of water samples.

Roger A. Newton

Roger A. Newton
Senior Nuclear Engineer

Subscribed and sworn to before me
this 1st day of November, 1978

Notary Public, State of Wisconsin

My commission expires 7/-14 1982

November 1, 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 the State of Wisconsin on October 2, 1978", were served upon those persons on the attached Service List by deposit in the United States mail, postage prepaid, this 1st day of November, 1978.

Roger A. Newton

Roger A. Newton
Senior Nuclear Engineer

Subscribed and sworn to before me
this 1st day of November, 1978.

Notary Public, State of Wisconsin

My commission expires 7-1-10 1982

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