SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE MODIFICATION OF THE SPENT FUEL STORAGE POOL

FACILITY OPERATING LICENSE NO. DPR-43

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

7902260489

1.0 INTRODUCTION

By letter and application dated November 14, 1977 and supplemented on March 13, 1978, July 10, 1978, August 18, 1978, September 5, 1978 and September 25, 1978 Wisconsin Public Service Corporation (WPSC) et al (the licensee) has requested an amendment to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. The request was made to obtain authorization to provide for additional storage capacity in the Kewaunee Spent Fuel Pool. The proposed modification would increase the capacity of the Spent Fuel Pool from the present capacity of 168 elements (of which 144 are located in the south pool and 24 in the north pool) to 990 elements (of which 720 would be located in the south pool and 270 in the north pool). The increased capacity would be achieved by installing new spent fuel storage racks with decreased spacing between fuel assembly storage slots. Present racks have a nominal center-to-center spacing between stored elements of 21 inches. The proposed spent fuel racks are double-walled stainless steel structures comprised of individual cavities which would provide a nominal center-to-center spacing of 10 inches between stored fuel elements. The general arrangement and details of the proposed new spent fuel storage racks are shown in the licensee's report "Spent Fuel Pool Modification Description and Safety Analysis" forwarded with the application for amendment dated Nove ber 14, 1977.

This Safety Evaluation addresses in addition to the results of our review of the proposed spent fuel pool modification, our evaluation of the impact of Lake Michigan faulting on the proposed facility modification. This evaluation is included as Appendix A to this Safety Evaluation Report.

2.0 DISCUSSION AND EVALUATION

2.1 Criticality Considerations

As stated in WPSC's November 14, 1977 submittal, the fuel pool criticality calculations are based on unirradiated fuel assemblies with no burnable poison and a fuel loading of 38.5 grams of uranium-235 per axial centimeter of fuel assembly. These calculations were made by the NUS Corporation for WPSC. The basic method was to use the NUMICE program, which is the NUS version of the LEOPARD program, to obtain four energy group cross sections for use in PDQ-7 diffusion theory calculations. The NUMICE program has "blackness theory" routines which were used to get the effective cross sections for the boron plates. These programs were used to calculate the neutron multiplication factor, k_∞ in the nominal lattice and then to calculate the change in k_∞ due to mechanical tolerances, changes in temperature, fuel and boron loading tolerances, missing boron plates, the eccentric loading of fuel assemblies in the storage locations, and a fuel assembly inadvertently positioned against the outside wall of a filled rack. NUS checked the accuracy of these diffusion theory calculations by making a KENO Monte Carlo calculation with 123 group cross sections which were obtained from the basic GAM-THERMOS library.

The calculated value for the maximum possible k_∞ for these fuel assemblies in the proposed racks is 0.901. If a fuel assembly is brought up against the outside of a filled rack, NUS calculated that the k_∞ could increase by 0.047. Thus NUS's calculated maximum worst case k_∞ is 0.948. Since the k_∞ is the neutron multiplication factor which is calculated by assuming no leakage of neutrons from the storage lattice, it is higher and thus more conservative than k_{eff}.

2.1.1 Evaluation

A comparison of the above results with the results of other calculations which were made for high density spent fuel storage lattices with boron plates shows them to be conservatively high. By assuming new, unirradiated fuel with no burnable poison or control rods, these calculations yield the maximum neutron multiplication factor that could be obtained during the life of the fuel assemblies. This includes the effect of the plutonium which is generated during the fuel cycle.

The NRC acceptance criterion for the criticality aspects of high density fuel storage racks, as stated in the staff's Branch Technical Position Review and Acceptance of Spent Fuel Storage and Handling Applications," is that the neutron multiplication factor, k_{eff}, in spent fuel pools shall be less than or equal to 0.95, including all uncertainties, under all conditions throughout the life of the racks. This 0.95 acceptance criterion is based on the overall uncertainties associated with the calculational methods, and it is our judgment that this provides sufficient margin to preclude criticality in fuel pools. Accordingly, there will be a Technical Specification which will limit the neutron multiplication factor, k_{eff}, in spent fuel pools to 0.95. In addition, in order to preclude any unreviewed increase, or increased uncertainty, in the calculated value of the neutron multiplication factor which could raise the actual k_{eff} in the fuel pool above 0.95 without being detected, a limit on the maximum fuel loading is also required. Accordingly, we find that the proposed high density storage racks will meet the NRC criteria when the fuel loading in the assemblies described in these submittals is limited to 38.5 grams or less of uranium-235 per axial centimeter of fuel assembly.

2.1.2 Conclusion

We find that when any number of the WPSC fuel assemblies which have no more than 38.5 grams of the uranium-235 per axial centimeter of fuel assembly are loaded into the proposed racks, the k_{eff} in the fuel pool will be less than the 0.95 limit. We also find that in order to preclude the possibility of the k_{eff} in the fuel pool from exceeding this 0.95 limit without being detected, it is necessary, pending further NRC review, to prohibit the use of these high density storage racks for fuel assemblies that contain more than 38.5 grams of uranium-235 per axial centimeter of fuel assembly. On the basis of the information submitted and the k_{eff} and fuel loading limits stated above we conclude that the proposed racks are acceptable with respect to criticality considerations.

2.2 Spent Fuel Cooling

The licensed thermal power for the Kewaunee Nuclear Power Plant is 1650 MWt. WPSC plans to refuel this plant annually. This will require the replacement of approximately forty of the 121 fuel assemblies in the core every year. In its November 14, 1977 submittal WPSC assumed a 112 hour decay time for calculating the maximum heat generation rates in the fuel pool for one third of a core, i.e., an annual refueling, and a 139 hour decay time for a full core offload. With these decay times WPSC used the ORIGEN program to calculate 19.0×10^6 BTU/hr as the maximum heat load for a full core offload that fills the pool with the proposed racks in place. This was assumed to take place one month after the startup following the 1997 refueling.

The spent fuel pool cooling system consists of two pumps and one heat exchanger. Each pump is designed to pump 450 gpm (2.25×10^5 pounds per hour) individually. When both pumps are operating with the single heat exchanger the design flow is 4.25×10^5 pounds per hour. With both pumps operating, the heat exchanger is designed to transfer 8.5×10^6 BTU/hr from 120°F fuel pool water to 66°F service water, which is flowing through the heat exchanger at a rate of 2.75×10^5 pounds per hour.

In its response to our request for additional information, WPSC stated that the Residual Heat Removal (RHR) heat exchanger would be available for cooling the spent fuel pool after a full core offload. It can be connected to the spent fuel pool cooling system by unbolting and turning spectacle flanges and opening isolation valves.

In its November 14, 1977 submittal WPSC states that "consistent with the structural and fuel element heat transfer analyses, the limiting condition for cooling system design and performance will be 150°F maximum bulk temperature with the failure of a single active component.

In its response to our request for additional information, WPSC stated that there are three safety class I sources of water for the spent fuel pool: a six-inch emergency service water supply line, a boric acid addition line, and a reactor water makeup line. Water from these lines can be delivered to the spent fuel pool by opening valves in existing lines. The largest of these lines, the emergency service water supply line, could supply pool makeup water at a rate of more than 1000 gpm.

2.2.1 Evaluation

Using the method given in Branch Technical Position ASB 9-2 of the NRC Standard Review Plan, with the uncertainty factor, K, equal to 0.1 for decay times longer than 10^3 seconds, we calculate that the maximum peak heat load during the twenty-fourth annual refueling could be 10.5×10^6 BTU/hr and that the maximum peak heat load for a full core offload that essentially fills the pool could be 22×10^6 BTU/hr. This full core offload was assumed to take place one year after the the twenty-first annual refueling. This assumption provides the maximum heat load. We also find that the maximum incremental heat load that could be added by increasing the number of spent fuel assemblies in the pools from 168 to 990 will be 2.8×10^6 BTU/hr. This is the difference in peak heat loads with full core offloads that essentially fill the present and the modified pools.

We calculate that with both SFP pumps operating, the spent fuel pool cooling system can maintain the fuel pool outlet water temperature below 133°F for a peak annual refueling heat load of 10.5×10^6 BTU/hr. We concur that the RHR system, needed for the full core offload situation, has sufficient cooling capacity when used in conjunction with the spent fuel pool cooling system to maintain a bulk pool temperature of 150°F. This 150°F is based on the core cooling for 100 hours before offloading is begun, with the entire unloading operation being completed in 39 hours. These times are delineated in the licensee's submittal of November 14, 1977.

Assuming a maximum fuel pool temperature of 150°F, the minimum possible time to achieve bulk pool boiling after any credible spent fuel pool cooling system failure will be about six hours. After bulk boiling commences, the maximum evaporation rate will be 46 gpm. We find that six hours would be sufficient time for WPSC to establish a 46 gpm makeup rate from makeup sources identified in Section 2.2. We also find that under bulk boiling conditions the surface temperature of the fuel will not exceed 350°F. This is an acceptable temperature from the standpoint of fuel element integrity and surface corrosion. It should be noted that because of redundant SFP cooling capability represented by the SFP cooling system and the RHR system, such a total loss of cooling would involve multiple single failures, an extremely unlikely situation.

2.2.2 Conclusion

We find that the present cooling capacity for the spent fuel pool at the Kewaunee Nuclear Power Plant will be sufficient to handle the incremental heat load including the increment that will be added by the proposed modifications. We also find that this total higher heat load will not alter the safety considerations of spent fuel cooling from those we previously reviewed and found to be acceptable. We conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the use of the proposed design with respect to adequate spent fuel pool cooling to accommodate the proposed modification.

2.3 Installation of Racks and Fuel Handling

There are two spent fuel pools at the Kewaunee Nuclear Power Plant. These are called the north pool and the south pool, and they are separated by about four feet of reinforced concrete. There are presently no spent fuel assemblies in the north pool and WPSC stated that there will be no fuel assemblies there during the installation of the proposed high density racks. This assumption is valid until the 1979 refueling outage, when, if the proposed modification is not yet completed, the licensee will have to store spent fuel in the north pool. After WPSC installs the high density storage racks in the north pool, it will use the present normal procedures to move the spent fuel assemblies that are in the south pool to the north pool. In this way, the racks in the south pool will also be changed without having spent fuel assemblies in the pool. Also WPSC states in its submittal that during the rack modification no components will be handled over spent fuel (Technical Specification 3.8.a.7). This will be assured by administrative procedures and by sight lines, barriers, crane stops, interlocks, and alarms as are determined to be necessary.

2.3.1 Evaluation

Since with the proposed administrative procedures there will be no fuel assemblies in the fuel pools undergoing the modification it will not be possible for an accident to result in any increased neutron multiplication factor. After the racks are installed in the pool, the boron in the absorber plates will afford protection against a criticality due to accidental deformation of the racks.

2.3.2 Conclusion

We conclude that there is reasonable assurance that there will be no increase in k_{eff} during relocation of spent fuel and related modification of the racks.

2.4 Structural and Mechanical Design

The proposed spent fuel pool modification consists of replacing the existing fuel storage racks in both the north and south pools with new spent fuel racks to eventually increase the storage capacity to a total of 990 fuel assemblies. Each of the new rack assemblies consists of a 9x10 rectangular array of stainless steel storage cells. The inner 56 storage positions are arranged on a 10 inch square pitch. The 34 storage positions in the peripheral rows are separated from the adjacent inner rows by 11 inches while the center-to-center spacing along the peripheral rows is maintained at 10 inches. Each storage cell consists of an 8.3 inch I.D. square stainless steel can, approximately 14 feet long, having a wall thickness of 0.125 inch with B_4C neutron absorber plates, supplied by Electroschmelzwerk Kempten (ESK) of West Germany, sealed within an annular gap between the can and an outer concentric can. The top and bottom of the annulus between cans will be closed with stainless steel seal rings and seal welded to provide a water-tight annulus within which the neutron absorber is held. A 0.25 inch diameter rod, tack welded along the length of each corner of the annulus, maintains the spacing between cans and provides lateral support for the absorber plates. Two stainless steel fuel supports, 1.25 inch X 1.25 inch, are welded along two sides of the bottom of the can.

All the rack assemblies will be bolted to support frame structures. These support frames are constructed of truss members with upper plates equipped with bearing plates and flow holes designed to mate with the fuel racks. Each support frame is designed to accept two rack assemblies, is rectangular in shape, and is supported by adjustable leveling legs that sit on the pool floor liner. The major trusses form a double rectangle with inner trusses across each rectangle to provide additional torsional rigidity. The frames are provided with adjustable seismic restraints which utilize the pool walls for support. Where two support frames meet they are bolted together. One of the two frames in the north pool is designed to accept only one rack as half of the frame will be in the equipment laydown area.

Further details of the racks and support frames are illustrated in the licensee's submittals.

The loads, loading combinations, and acceptance criteria are in accordance with Section 3.8.4 of the Standard Review Plan. The allowable stresses for both the type 304 and 17-4 PH stainless steel are in accordance with Section III of the ASME Boiler and Pressure Vessel Code. The allowable stresses for the stainless steel welds are as specified in Table NF-3292.1-1 of the ASME Code. The yield strengths for the SA-240 type XM-29 and the ASTM A-276 type UNS-210-800 stainless steels are from the ASTM Material Specifications and are adjusted for temperature using data provided by the material supplier.

The seismic analysis performed was a modal response spectrum analysis using 1.0% damping for both OBE and SSE. Loads, stresses and deflections were determined for a group of four rack assemblies and support frames in the most conservative horizontal direction. The results of this analysis were then combined with the response in the vertical direction by the absolute sum method. All water inside the cans and surrounding the fuel and the water surrounding the cans themselves is added to the mass of the racks. Fuel weight is accounted for in both the frequency and load calculations for the linear analysis. The fuel mass is again included in the overall rack/support frame analysis when the response from a non-linear analysis of fuel assembly/can impacting is combined with the linear response spectrum analysis.

The racks have been designed to withstand the local as well as gross effects of a dropped fuel assembly. Straight and inclined drops on the lead-in guides on top of the cans were considered as well as drops directly through cans in both a flexible location and over one of the leveling legs. Results of impact testing from an article entitled, "Plastic Impact Testing of Shipping Cask Fin Specimens," by F. C. Davis and H. Pik, were referenced as a basis for part of the analyses.

The effects from a postulated stuck fuel assembly have been examined assuming a maximum uplift load of 4000 lbs. (capacity of the crane).

Because of the increased loading imparted to the pool resulting from this increase in storage capacity, a structural analysis was made of the pool walls and floor. The load combinations considered were per Standard Review Plan Section 3.8.3.II.3 and the allowable loads were taken from the ACI 318-63 Code.

All rack and support frame components, as discussed previously, are fabricated of stainless steel. The 17-4 PH stainless steel being utilized will be heat treated to at least 1100°F, the surface film removed by either pickling or grit blasting, and correct heat treatment verified by destructive examination of test samples heat treated along with each lot of material.

The new racks will be installed on a phased basis. The existing racks in the north pool, which contains no fuel, will be removed and new support frames and rack assemblies installed. All fuel from the south pool will then be transferred to the north pool. The existing racks in the south pool will then be removed and support frames for all eight new racks and four new racks, will be installed. The remaining four racks will be installed as needed.

2.4.1 Evaluation

The design, fabrication, and installation procedures, the structural design and analyses procedures for all loadings, (including seismic and impact loading), the load combinations and structural acceptance criteria, the quality control for the design, fabrication, and installation, and the applicable industry codes were all reviewed in accordance with the Branch Technical Position (BTP) entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications."

One of the acceptance criteria presented in the BTP is the use of Regulatory Guide 1.92 methods for combining earthquake responses. However, the licensee has used the absolute sum method to combine the response from one horizontal with the vertical response. In order to show conformance with Regulatory Guide 1.92, which finds the SRSS method of combining all three responses acceptable, WPSC has done an analysis that shows, for the resultant stresses obtained from the seismic analysis, that combining one horizontal response with the vertical response by the absolute sum method is as conservative as combining both horizontal responses with the vertical response by the SRSS method when the mass of the fuel assemblies is only included in the non-linear analysis. Also, a conservatism is included in the analysis since the fuel weight was included in the frequency calculations. This resulted in a lower fundamental frequency and correspondingly higher value of acceleration than would have resulted if the frequencies of the rack assemblies alone were utilized.

Results of the seismic analyses considering the most conservative arrangement of rack assemblies that will exist in the pool show that the racks and support frames are capable of withstanding the loads associated with all the design loading conditions without exceeding allowable stresses. Also, impact due to fuel/can interaction will result in no damage to the racks or fuel assemblies themselves.

Results of the dropped fuel assembly analyses show that local deformation will occur, but indicate that gross stresses meet the applicable allowables.

Procedures to preclude the impact of heavy loads on spent fuel during rack installation are addressed in Section 2.7, Fuel and Heavy Load Handling.

Results of the stuck fuel assembly analysis show that stresses are below those allowed for the applicable loading combination.

Results of the structural analysis of the pool show that the present load carrying capacity of the pool is adequate.

The neutron absorber plates are being supplied by Electroschmelzwerk Kempten (ESK) of West Germany because of the off-gassing problem experienced with domestically fabricated B_4C plate material. Testing indicates that exposure to radiation results in no measurable decrease in strength. Also, results of the testing to date show that no significant off-gassing of the material occurs.

Since the possibility of long term storage of spent fuel exists, the effects of the pool environment on the racks and fuel cladding must be examined. The rack assemblies and support frame components are all stainless steel. Operating experience indicates that at the pool temperature and the quality of the demineralized water (with dissolved boric acid), it is highly unlikely that the racks, support frame or fuel cladding will incur any corrosion problems during the life of the plant. Also, corrosion of the B_4C neutron absorber plates will not be a problem. The material is sealed within the cans and all seal welds dye penetrant inspected prior to rack assembly.

All racks will be seismically supported throughout all construction phases and no components will be handled over spent fuel during the changeout operation.

2.4.2 Conclusion

Based on the evaluation presented above, we find that the new proposed Kewaunee spent fuel storage racks and the design and analyses performed for the racks, support frames, and pool are in conformance with established criteria, codes and standards specified in the staff position for acceptance of spent fuel storage and handling applications and satisfies the applicable requirements of the General Design Criteria 2, 4, 61 and 62 of 10 CFR, Part 50, Appendix A.

We find the modification proposed by the licensee to be structurally and mechanically acceptable.

2.5 Occupational Radiation Exposure

We have reviewed the licensee's plans for removal and disposal of the old low density racks and the installation of the new high density racks with respect to occupational exposure. The new high density racks will be installed in the spent fuel pool in two steps. Seven racks will be installed in 1979 and four racks in the 1980's. All the old low density racks will be removed from the pool and disposed of during the first step of the modification. All the support structure for the new high density racks will be installed by divers in the north and south pools during the first step of the modification The spent fuel in the SFP will be in the south pool when the divers are working in the north pool and vice versa. Divers will not be used during the second step of the modification.

In the matter of disposal of the old low density racks, WPSC is considering two alternative plans: crating and shipping the racks intact versus cutting, crating and shipping the racks. The licensee has submitted an analysis of the occupational exposure for the first step of the pool modification with the old racks being cut into smaller sections to permit more efficient packaging in the shipping containers. More efficient packing results in a smaller volume of radioactive waste to be disposed of with resulting economic and environmental benefits, e.g., fewer waste shipments and conservation of low level waste burial site space. This option, however, does require that the licensee expend efforts to cut the old racks and results in a slight increase in occupational radiation exposure. The occupational radiation exposure for the first step of the pool modification with cutting, crating and shipping the racks has been estimated by the licensee to be 11.6 man-rem. WPSC has not estimated the occupational exposure for the pool modification with crating and shipping the racks intact but this exposure will be less than the estimated 11.6 man-rem for cutting the racks. Based on the licensee's estimate of occupational exposure for the SFP modification, we would estimate the occupational exposure for the SFP modification with crating and shipping the racks intact to be about 9.6 manrem. WPSC has not yet quantified a cost-benefit analysis of the alternatives so that their disposal decision has not been finalized. In any event, WPSC will base their decision on this cost-benefit analysis of the alternatives so that exposures will be kept to levels that are as low as is reasonably achievable (ALARA).

Installing the new high density racks in the pool in two steps instead of completing the modification in a single step is acceptable because the occupational exposure for either method of installation should be approximately the same. The south pool is contaminated from three refuelings. The proposed modification is not expected to significantly increase the pool water activity and resulting radiation levels in the vicinity of the pool. Divers will not be needed during the installation of the last four racks. Therefore, the occupational exposure for installing the new racks in two steps should be approximately the same as for installing these racks in a single step. Based on the licensee's estimate of the occupational exposure to install four new racks in the south pool during the first step of the spent fuel pool modification, we have estimated an additional 0.4 man-rem for the completion of the second step of the pool modification. The licensee will not have to dispose of any old low density racks during this second step of the modification.

The occupational radiation exposure for both steps of the pool modification is estimated to be about 12 man-rem. This represents a small fraction (about 0.2%) of the total man-rem burden from occupational exposure at the plant during its lifetime.

We have estimated the increment in the annual onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by WPSC and by utilizing relevant assumptions for occupancy times and for dose rates in the spent fuel area from radionuclide concentrations in the SFP water. The spent fuel assemblies themselves contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the proposed action represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than two percent to the total annual occupational radiation exposure burden at this facility. The small increase in radiation exposure will not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable and within the limits of 10 CFR 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

2.6 Radioactive Waste Treatment

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. The waste treatment systems were evaluated in the Safety Evaluation Report (SER) for the Kewaunee Nuclear Power Plant dated July 1972. There will be no change in the waste treatment systems or in the conclusions of the evaluation of these systems as described in Section 11.0 of the SER because of the proposed modification since there will be no significant increase in radioactive waste.

2.7 Fuel and Heavy Load Handling

The NRC staff has underway a generic review of load handling operations in the vicinity of spent fuel pools to determine the likelihood of a heavy load impacting fuel in the pool and, if necessary, the radiological consequences of such an event. Kewaunee currently has a Technical Specification (TS 3.8.a.7) which does not allow heavy loads greater than the weight of a fuel assembly to be transported over or placed in either part of the SFP when spent fuel is stored in that part. The licensee plans to install the new high density racks in two steps. During the second phase of rack installation, placement of new racks will be permitted only if the racks do not traverse directly above spent fuel stored in either the north or south pool.

We have concluded that the likelihood of a heavy load handling accident is sufficiently small that the proposed modification is acceptable.

The consequences of fuel handling accidents in the spent fuel pool area are not changed from those presented in the Safety Evaluation Report (SER) dated July 1972.

3.0 SUMMARY

Our evaluation supports the conclusion that the proposed modification to the Kewaunee SFP is acceptable because:

(1) The increase in occupational radiation exposure to individuals due to the storage of additional fuel in the SFP would be negligible.

(2) The installation and use of the new fuel racks does not alter the consequences of the design basis accident for the SFP, i.e., the rupture of a fuel assembly and subsequent release of the assembly's radioactive inventory within the gap.

(3) The likelihood of an accident in[illegible]g heavy loads in the vicinity of the spent fuel pool [illegible]ficiently small.

(4) The physical design of the new storage racks will preclude criticality for any credible moderating condition with the limits to be stated in the Technical Specifications.

(5) The SFP has adequate cooling with existing systems.

(6) The structural design and the materials of construction are adequate to function normally for the duration of plant lifetime and to withstand the seismic loading of the design basis earthquakes.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and that the proposed action to permit installation and use of high density spent fuel storage racks in the spent fuel pool at the Kewaunee Nuclear Power Plant, will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 1, 1978

APPENDIX A

IMPACT OF LAKE MICHIGAN FAULTING ON PROPOSED SPENT FUEL POOL MODIFICATION KEWAUNEE NUCLEAR POWER PLANT

On June 22, 1978, during a visit to the Haven site, Wisconsin Electric Power Company presented to the NRC staff preliminary geologic information on NNE-training faults within Lake Michigan. These data were presented as an initial response to NRC round one questions. Sufficient information was not presented at that time to define the faults' characteristics. An amendment to the Haven PSAR on the geology of Lake Michigan is currently being reviewed by the NRC staff. The applicant has stated that additional studies of the faults are being conducted and will be included in a future amendment to the Haven Preliminary Safety Analysis Report.

On August 23, 1978, the staff informed the Kewaunee Atomic Safety and Licensing Board of the preliminary information received on the Haven docket and referred to herein. The staff indicated that our safety evaluation relevant to the proposed spent fuel pool modification at the Kewaunee Nuclear Power Plant would address the significance, if any, of additional geological information to the proposed facility modification. Our evaluation follows.

Based on the tectonic history of the region and the absence of historic seismicity, we have a high degree of confidence that the faults beneath Lake Michigan are geologically old and pose no potential to increase the earthquake hazard of the region. The Haven site is located on the western edge of the Michigan Basin within the Central Stable Region tectonic province. This province is generally characterized by gentle arches, domes and basins (i.e., Michigan Basin) which formed during several tectonic epeirogenic episodes (episodes of broad gentle vertical movement of the Earth's crust) during the Paleozoic Era more than 225 million years ago (mybp). There is no known geologic evidence of tectonic deformation or faulting in the region subsequent to that time. Faulting within the Paleozoic age rocks in the Central Stable Region was, however, widespread prior to and including the deposition of the Mississippian age rocks (320 + mybp). The discovery of faulting within Mississippian rock units beneath Lake Michigan was, therefore, not unexpected. On the contrary it is consistent with the known tectonic history of the region.

Based on the information available to the staff at the present time, we do not consider the indications of faulting near the Haven site to be relevant and material to previous staff conclusions with respect to the geologic hazard at Kewaunee.

In view of the above, we recommend that the licensing action associated with the proposed Kewaunee spent fuel pool modification not be delayed pending submittal and review of additional information on faulting near the Haven site.

DL

R. 2
Sheboygan, Wisc.
November 6, 1978

Re: amendment 15: utilities plan to store radioactive wastes on Lake Michigan near Sheboygan.

Dear senator,

We are pleading for your help, so far the government has not found a suitable place to dump the spent fuel from nuclear plants, now the utilities say they want to store the spent fuel at the Haven site, we don't approve of that, neither do we approve of the planned nuclear plant, we'd have that radiation belcher practically in our back yard. It would be 1 mile across fields from us, but then there would be the high power poles. Did you know that if you walked under one of these it changes the molecules in your body, and when a farmer drives underneath with his tractor, it shorts out? Then they want to stick us with burying the waste!! Now they talk about building one at Port Washington too, we'll have nothing but a radiation belt along Lake Michigan. The life span of these plants is too short (30) years, then what? If they would give the coal miners a fair shake, get rid of the corruption in their unions it would be a whole lot better.

We hope you can help resolve this problem of nuclear waste disposal, so far the government hasn't been able to do a thing.

Sincerely,

Albert and Helen Wiedemann

Since publication of the draft NUREG in September, 1977, the Commission directed the staff to reevaluate the long-term impact of radon-222 from the uranium fuel cycle. The reevaluations have been included in the Perkins,Pebble Springs and Black Fox Hearings records in May and June, 1978. Health effects estimates from radon have been conservatively extended into an admittedly uncertain future to incorporate periods ranging from 100 to 1,000 years. Similarly, the staff also extended health effects estimates of carbon-14 releases for 100 to 1,000 years into the future.

These estimates have now been incorporated into the comparison of health effects for the coal and nuclear fuel cycles. The revised tables and Summary and Conclusion sections of the draft NUREG are attached.

In addition, some believe (Ref. 33) that when the physical and biological properties of the radium released from conventional coal powered plants burning coal (with 1-2 ppm uranium-238 and Th-232) are considered, such plants discharge relatively greater quantities of radioactive materials into the atmosphere than nuclear powered plants of comparable size. EPA has estimated radiation doses from coal and nuclear powered plants of early designs and reached similar conclusions (Ref. 16). Even if the health effects from radioactivity released by the coal fuel cycle are greater than the health effects from radioactivity released in the nuclear fuel cycle, the total health effects from coal would not change significantly since these effects would be only a small percentage of the total health effects from the coal cycle.

III. SUMMARY AND CONCLUSIONS

For the reasons discussed above, it is extremely difficult to provide precise quantitative values for excess mortality and morbidity, particularly for the coal fuel cycle. Nevertheless, estimates of mortality and morbidity have been prepared based on present day knowledge of health effects, and present day plant design and anticipated emission rates, occupational experience and other data. These are summarized in Tables 1 and 2, with some important assumptions inherent in the calculations of health effects listed in Appendix A.

While future technological improvements in both fuel cycles may result in significant reductions in health effects, based on current estimates for present day technology, it must be concluded that the nuclear fuel cycle is considerably less harmful to man than the coal fuel cycle. (Refs. 1,2,3, 4,5,10,11,27,28,33,34,35,36) As shown in Tables 1 and 2, the coal fuel cycle alternative may be more harmful to man by factors of 7.3 to 42 depending on the effect being considered, for an all nuclear economy, or factors of 5.5 to 14 with the assumption that all of the electricity used by the uranium fuel cycle comes from coal powered plants.

It should be noted that although there are large uncertainties in the estimates of most of the potential health effects of the coal cycle, the impact of transportation of coal is based on firm statistics; this impact alone is greater than the conservative estimates of health effects for the entire uranium fuel cycle (all nuclear economy), and can reasonably be expected to worsen as more coal is shipped over greater distances. In the case where coal generated electricity is used in the nuclear fuel cycle, primarily for uranium enrichment and auxiliary reactor systems, the impact of the coal power accounts for essentially all of the impact of the uranium fuel cycle.

Table 1. Current Energy Source Excess Mortality Summary per Year per 0.8 GWy(e)

	Occupational		General Public		Totals
	Accident	Disease	Accident	Disease	
Nuclear Fuel Cycle (all nuclear)	0.22 (a)	0.14 (b)	0.05 (c)	0.18-1.3 (b)	0.59-1.7 (1.0)*
(with 100% of electricity used in the fuel cycle produced by coal power (U.S. population for nuclear effects; regional population for coal effects)	0.24-0.2[illegible] (d)	0.14-0.46 (b,e)	0.10 (c,f)	0.77-6.3 (g)	1.2-6.8 (2.9)
Coal Fuel Cycle (Regional Population)	0.35-0.65 (d)	0-7 (e)	1.2 (f)	13-110 (g)	15-120(42)

Ratio of Coal to Nuclear: (geometric means)
42 (all nuclear)
14 (with coal power)(h)

(a) Primarily fatal non-radiological accidents such as falls, explosions, etc.
(b) Primarily fatal radiogenic cancers and leukemias from normal operations at mines, mills, power plants and reprocessing plants.
(c) Primarily fatal transportation accidents (Table S-4, 10 CFR 51) and serious nuclear accidents.
(d) Primarily fatal mining accidents such as cave-ins, fires, explosions, etc.
(e) Primarily coal workers pneumoconiosis (CWP) and related respiratory diseases leading to respiratory failure..
(f) Primarily members of the general public killed at rail crossings by coal trains.
(g) Primarily respiratory failure among the sick and elderly from combustion products from power plants, but includes deaths from waste coal bank fires.
(h) 100% of all electricity consumed by the nuclear fuel cycle produced by coal power; amounts to 45 MWe per 0.8 GWy(e).
* Values in parentheses are the geometric means of the ranges; geometric mean = $\sqrt{ab}$

Table 1a
(Breakdown of Table 1)

NUCLEAR — EXCESS MORTALITY per 0.8 GWy(e)

FUEL CYCLE COMPONENT	OCCUPATIONAL		GENERAL PUBLIC		TOTAL
	ACCIDENT (a)	DISEASE (b,c,d,)	ACCIDENT (d,e,)	DISEASE (b)	
RESOURCE RECOVERY (Mining, Drilling, etc.)	0.2	0.038	~0	0.085+	
PROCESSING (f)	0.005**	0.042	*	0.026-1.1 (g)	
POWER GENERATION	0.01	0.061	0.04	0.016-0.20	
FUEL STORAGE	*	~0	*	~0	
TRANSPORTATION	~0	~0	0.01	~0	
REPROCESSING	*	0.003	*	0.054-0.062	
WASTE MANAGEMENT	*	~0	*	0.001	
TOTAL	0.22	0.14	0.05	0.18-1.3	0.59-1.7

+These effects are based on my affidavit of March 28, 1978 which indicates that the 4,060 Ci of Rn-222 released from mining the uranium necessary to produce the 0.8 GWy(e) would result in 0.085 excess deaths over all time.

*The effects associated with these activities are not known at this time. While such effects are generally believed to be small, they would increase the totals in this column.

**Corrected for factor of 10 error based on referenced value (WASH-1250)

(a) Ref. 1
(b) Ref. 7
(c) 10 CFR 51, Table S-3
(d) 10 CFR 51, Table S-4
(e) Ref. 8
(f) Includes milling, uranium hexaflouride production, uranium enrichment, and fuel fabrication.
(g) Long-term effects given by affidavit March 28, 1978 for radon-222 releases from mills and tailings piles account for all but 0.001 health effects here.

Table 2. Current Energy Source Summary of Excess Morbidity and Injury per 0.8 GWy(e) Power Plant

	Occupational		General Public		Totals
	Morbidity	Injury	Morbidity	Injury	
Nuclear Fuel Cycle (all nuclear)	0.84 [a]	12 [b]	1.0-3.1 [c]	0.1 [d]	14-16 (15)*
(with 100% of electricity used by the fuel cycle produced by coal power) (U.S. population for nuclear effects; regional population for coal effects)	1.7-4.1 [e]	13-14 [b]	1.5-7.6 [g]	0.55 [h]	17-26 (21)
Coal Fuel Cycle (Regional population)	20-70 [e]	17-34 [f]	10-100 [g]	10 [h]	57-210 (109)

Ratio of Coal to Nuclear: (geometric means) 7.3 (all nuclear)

5.2 (with coal power)[i]

(a) Primarily non-fatal cancers and thyroid nodules.
(b) Primarily non-fatal injuries associated with accidents in uranium mines such as rock falls, explosions, etc.
(c) Primarily non-fatal cancers, thyroid nodules, genetically related diseases, and non-fatal illnesses following high radiation doses such as radiation thyroiditis, prodromal vomiting, and temporary sterility.
(d) Transportation related injuries from Table S-4, 10 CFR Part 51.
(e) Primarily non-fatal diseases associated with coal mining such as CWP, bronchitis, emphysema,etc.
(f) Primarily injuries to coal miners from cave-ins, fires, explosions, etc.
(g) Primarily respiratory diseases among adults and children from sulfur emissions from coal-fired power plants, but includes waste coal bank fires.
(h) Primarily non-fatal injuries among members of the general public from collisions with coal trains at railroad crossings.
(i) 100% of all electricity consumed by the nuclear fuel cycle produced by coal power; amounts to 45 MWe per 0.8 GWy(e).
* Values in parentheses are the geometric means of the ranges.

ENCLOSURE 5

Table 2a
(Breakdown of Table 2)

NUCLEAR — MORBIDITY AND INJURY per 0.8 GWy(e)

FUEL CYCLE COMPONENT	OCCUPATIONAL		GENERAL PUBLIC		TOTAL
	MORBIDITY	INJURY (a)	MORBIDITY	INJURY (b)	
RESOURCE RECOVERY (Mining, Drilling, etc.)	**	10	***	~0	
PROCESSING (c)	**	0.6	***	~0	
POWER GENERATION	**	1.3	***	~0	
FUEL STORAGE	**	*	***	~0	
TRANSPORTATION	**	<1	***	0.1	
REPROCESSING	**	*	***	*	
WASTE MANAGEMENT	**	*	***	~0	
TOTAL	0.84	12	1.0-3.1	0.1	14-16

(a) Ref. 1
(b) Table S-4, 10 CFR 51
(c) Includes milling, uranium hexaflouride production, uranium enrichment, and fuel fabication.

*The effects associated with these activities are not known at this time. While such effects are generally believed to be small, they would increase the totals in this column.

**Non-fatal cancers $\leq$ fatal cancers (excluding thyroid) = 0.14
Non-fatal thyroid cancers and benign nodules = 3X fatal cancers = 0.42
Genetic defects 2X fatal cancers = 0.28

***Reactor accidents 10X fatalities = 0.40 non-fatal cases
Normal operations: Non-fatal cancers $\leq$ fatal cancers = 0.18-1.3
Non-fatal thyroid cancers and nodules = 3X fatal cancers (from Total-Body doses) = 3X(0.085-0.28) = 0.26-0.84
Genetic Effects = 2X fatal cancers (from Total-Body doses) = 2X(0.085-0.28) = 0.17 - 0.56