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1.0 Introduction

The existing fuel enrichment limit for the Kewaunee Nuclear Power Plant (KNPP) is 38.5 grams (3.67 w/o) of uranium-235 per axial centimeter of fuel assembly. This limit was established in 1978 based on the on-site storage capacity and the expectation that a viable nuclear fuel reprocessing and/or disposal program would be in place by the mid-1990s. At that time, it was also planned to maintain enough storage space in the spent fuel pool for an unscheduled removal of an entire core.

The primary benefit of increased fuel enrichments is the resulting fuel management flexibility. This flexibility will aid in the implementation of the KNPP reactor vessel flux reduction program. Reducing the reactor vessel flux will extend the useful life of the reactor vessel and facilitate potential life extension and license renewal.

Also, continuing plant operation with the current fuel enrichment levels will result in the depletion of current on-site spent fuel storage capacity by 1998 if the capability to store one full core offload is maintained. Given the unsettled state of the high level nuclear waste disposal program, we are looking to extend the life of our existing on-site storage capability. Increasing the enrichment level will reduce the fuel assembly feed batch sizes, which will in turn reduce the annual demand on the ever decreasing number of available spent fuel pool storage locations.

ATTACHMENT 1

Proposed Amendment Request 95
to the
Kewaunee Nuclear Power Plant
Technical Specifications

July 5, 1990

Evaluation of the Use of the Higher Fuel
Enrichments at the Kewaunee Nuclear Power Plant

Approval of this amendment request would increase the fuel enrichment limit to 49.2 grams (4.75 w/o) of uranium-235 per axial centimeter. The analyses in Attachment 3 demonstrate that the supporting systems can accommodate this new limit and that NRC k-effective criteria, as defined in NUREG-0800, Rev. 2, will be met without additional administrative controls on new and spent fuel storage locations. Contingent on NRC approval of this change, Wisconsin Public Service Corporation (WPSC) will implement a gradual increase above the current fuel enrichment starting with cycle 18 (1992). It is projected that fuel enrichments up to approximately 4.2 w/o will be in use at KNPP by 1997. This gradual increase in fuel enrichment will add approximately two years to the current on-site storage capability. An approved limit of 4.75 w/o would allow flexibility to extend on-site storage capabilities even further, if necessary.

This report and the other attachments to this letter discuss the various aspects which were evaluated in support of this change and demonstrate that a KNPP fuel enrichment limit of 49.2 grams (4.75 w/o) of uranium-235 per axial centimeter will have no detrimental effect on the health and safety of the public.

2.0 Description of the Proposed Change

It is proposed to add a paragraph, 2.C.(7), to the Kewaunee Nuclear Power Plant (KNPP) operating license, DPR-43, to allow the reload of fuel assemblies with enrichments up to 49.2 grams of uranium-235 per axial centimeter and the storage of such assemblies prior to and subsequent to loading in the reactor. The current fuel enrichment limit of 38.5 grams of uranium-235 per axial centimeter is not specified in the KNPP Technical Specifications or Operating License. The current limit was established in the NRC's SER for the use of high density storage fuel storage racks in the spent fuel pool at the KNPP, dated December 1, 1978.

Criticality analyses were performed by B&W Fuel Company in support of this proposed change. A copy of the criticality analysis report can be found in Attachment 3. It is anticipated that enrichments above the current limit of 38.5 grams of uranium-235 per axial centimeter will be incorporated into the KNPP cycle 18 (1992) core design.

3.0 General Description of the Design and Functions of Support Systems

KNPP is a 560 MWe PWR (Westinghouse) with a total of 121 fuel assemblies in the reactor core. Spent fuel pool storage capacity consists of two pools, "North" and "South", which are connected to each other and have a common Spent Fuel Pool Cooling System. Water in the spent fuel pool is maintained at a minimum concentration of 2100 ppm boron. The basic components of the Spent Fuel Pool Cooling System are two pumps, two filters, one heat exchanger, one cleanup demineralizer and associated valves, piping and instrumentation.

The Spent Fuel Pool Cooling System was evaluated and determined to have heat removal capability for 22 normal refuelings plus a complete core unload after 100 hours decay time. Under those conditions the pool would be maintained at or below 150°F. With the postulated failure of a single active component (such as a Spent Fuel Pool pump) there would be 1.5 days to repair or replace the component before 150°F would be exceeded with the pool loading conditions described. Therefore, since there is a large capacity for heat absorption in the spent fuel pool, active system components are not redundant. Alternate cooling capability can be made available through existing interconnections with the Residual Heat Removal System, if necessary due to malfunctions or failures. The Spent Fuel Pool Cooling System also contains provisions for filtering and demineralization, to minimize radioactivity levels in the system.

The KNPP spent fuel storage capacity consists of high density storage racks with 990 locations for fuel assemblies. Of these locations 556 have already been filled. The Spent Fuel Storage Racks (SFSR) are designed to store new fuel and irradiated fuel in a vertical configuration under water. The safety functions of the SFSR are to maintain the fuel assemblies in a non-critical configuration and to prevent damage to the fuel assemblies during a seismic event or a dropped assembly event. The racks were designed with a nominal assembly pitch of 10 inches. The SFSR cans contain boron carbide plates which are neutron absorbers. The rack and assembly spacings, along with the boron carbide plates, ensure a k-effective of less than 0.95 when the SFSR is immersed in water.

The KNPP new fuel storage capacity consists of storage racks with 44 locations for fuel assemblies. The New Fuel Storage Racks (NFSR) store new, non-irradiated nuclear fuel assemblies in a dry, vertical configuration.

The safety functions of the NFSR are to maintain the fuel assemblies in a non-critical configuration and to prevent damage to the fuel assembles during a seismic event. The racks were designed with a minimum center-to-center spacing of 20.24 inches between fuel assemblies. This spacing ensures a k-effective of less than 0.98 for misted conditions and less than 0.95 if the entire rack is flooded with water. The new fuel assembly storage cells are an "open air" design with stainless steel bracing at the top, bottom, and midway point of the rack.

The support systems had been previously evaluated for fuel enrichments up to 38.5 grams (3.67 w/o) of uranium-235 per axial centimeter of fuel assembly. Those evaluations conservatively demonstrated conformance with design requirements and compliance with NRC k-effective limits. Evaluations were recently performed again based on fuel enrichments up to 49.2 grams (4.75 w/o) of uranium-235 per axial centimeter of fuel assembly in support of this change. These new evaluations demonstrated that all of the support systems can adequately handle any additional loads resulting from this increased fuel enrichment.

4.0 Criticality Analyses Summary for Higher Enrichments

Criticality analyses were performed for the Spent Fuel Storage Racks (SFSR) and the New Fuel Storage Racks (NFSR) to determine the most limiting condition. The NFSR resulted in the most limiting fuel enrichment of 49.2 grams (4.75 w/o) of uranium-235 per axial centimeter of fuel assembly. The analyses performed also demonstrated the following for the requested enrichment limit:

1. No administrative controls, such as burnup credit, are required.

2. No criticality concern exists for the SFSR when flooded with unborated water.

3. No criticality concern exists for the NFSR when flooded with unborated water or with optimally misted moderator conditions.

Abnormal occurrences were included in the criticality analysis. Occurrences such as off-center fuel placement, T-bone assembly drop accidents, and assemblies misplaced between the racks and spent fuel pool walls (misplaced assembly accidents) were evaluated. Assemblies misloaded within the SFSR and NFSR were not considered because all fresh fuel at the maximum enrichment was assumed, thereby eliminating this type of accident from further consideration.

The following summarizes the assumptions and results of the SFSR and NFSR criticality analyses.

4.1 SFSR Criticality Analysis

Criticality analyses were performed for the SFSR to determine the maximum fuel enrichment which would still maintain k-effective $\leq$ 0.95. These analyses resulted in a calculated maximum enrichment of 52.3 grams (5.05 w/o) of uranium-235 per axial centimeter.

4.1.1 SFSR Criticality Analysis Assumptions

The calculational models assumed the following conditions for the SFSR analyses:

a) No structural braces or material were considered for the rack except the inner and outer stainless steel 304 cans and B_4C poison plates.

b) No soluble poisons were considered. This condition is required by ANSI/ANS-57.2-1983 Section 6.4.2.2.9 except for evaluation of plant condition (PC) IV and V faults.

c) In the enrichment studies for fresh fuel all water was assumed to be at the optimum temperature of 50°F.

d) In some calculations the racks were assumed to be infinite in the radial X-Y directions and have at least a twelve inch 100% dense water reflector above the design length assembly. This is a conservative assumption because fuel is more reactive than a source of reflected neutrons from the concrete walls. Other rack calculations modeled the specific dimensions of the SFSR.

e) For conservatism all fuel contained an enrichment tolerance of +0.05 wt% U^{235}.

f) The fuel assemblies contained no burnable poisons. This assumption is required because ANSI/ANS-57.2-1983, section 6.4.2.2.8 indicates that credit for poisons may only be taken for inherent structural materials or for poisons that cannot be accidentally removed by mechanical or chemical action. Since burnable poison clusters are not an integral part of the structural assembly design and can be mechanically removed, credit may not be taken for them.

g) Maximum thicknesses were used for the stainless steel inner and outer cans that surround the B_4C poison plates. Maximum thicknesses were used to provide a physical means to reduce the thickness of the water flux trap between cans. The reactivity effect of increased absorption of neutrons by steel is very small compared to flux trap effects and is also offset by neutron reflection from the inner steel can back to the fuel assembly.

h) Minimum assembly pitches were calculated using the cumulative tolerances in the X-direction and Y-directions. These assembly pitches were specifically evaluated in the off-center spacing studies.

i) Two pool designs are present in the SFSR. The larger pool design consists of a 2X4 array of 9X10 assembly racks while the smaller pool consists of a 2X2 rack area but contains only three 9X10 assembly racks. The unused area is the shipping cask laydown area. The most reactive part of the SFSR is the larger pool area since it contains the greatest volume of fuel, so it was assumed to be representative of the SFSR as a whole.

j) No intermediate spacer grids were modeled.

k) No seismic events were considered since the SFSR is Class I seismically rated and changing the assembly enrichment does not change the racks' structural characteristics or tolerances.

4.1.2 SFSR Criticality Analysis Results

The criticality analysis results demonstrated that the spent fuel pool is critically safe for all normal and accident conditions with fuel enrichments up to and including 52.3 grams (5.05 w/o) of uranium-235 per axial centimeter of fuel assembly. The limiting SFSR accident was found to be the spent fuel pool flooded with unborated water concurrent with a misplaced assembly. The maximum k for that scenario, considering uncertainties and biases, was calculated to be 0.93428 with a 95/95 tolerance level. This is well within the NRC imposed k-effective limit of 0.95.

4.2 NFSR Criticality Analysis

Criticality analyses were performed for the NFSR to determine the maximum fuel enrichment which would still maintain k-effective $\leq$ 0.95 for flooded conditions and $\leq$ 0.98 for misted conditions. These analyses resulted in a calculated maximum enrichment of 49.2 grams (4.75 w/o) of uranium-235 per axial centimeter.

4.2.1 NFSR Criticality Analysis Assumptions

The calculational models assumed the following conditions for the NFSR:

a) No structural braces or material were considered for the rack except the upper and lower part length assembly guides and the 0.25 inch stainless-steel cover plates. For the flooded cases the part length assembly guides were either approximated by a stainless-steel can or the guides were eliminated. For interspersed moderator conditions the part length assembly guides were modeled as Zircaloy-4 cans or were eliminated. Zircaloy-4 was used instead of stainless steel since it would provide the neutron reflection of steel but not the neutron absorption and is therefore a conservative approximation.

b) No soluble poisons were considered for flooded cases since it is possible that poison could be removed by mechanical or chemical action. This condition is required by ANSI/ANS-57.2, Section 6.4.2.2.9 and ANSI/ANS-57.3-1983, Section 6.2.4.2.

c) In the criticality studies for fresh fuel the most reactive temperature was determined. The limiting accident cases were evaluated under optimum moderator conditions.

d) For the 100% dense water moderated cases the racks were assumed to be infinite in the radial X-Y directions. This is a conservative assumption because fuel is more reactive than a source of reflected neutrons from the concrete walls. At least a twelve inch water reflector is modeled above the design length assembly and aluminum covers and is sufficient for the system to be considered decoupled from the thermal neutron spectrum. A 24 inch thick concrete floor was modeled underneath the NFSR. The 1-5/8 inch plywood flooring was conservatively modeled as water.

e) For the misted or fogged condition, 1/4 of the rack is modeled with its nearby concrete walls to determine the worst case moderator conditions. For accident cases 1/2 of the rack was modeled. A thickness of at least 13 feet of low-density moderator was assumed above the active fuel height.

f) All fuel contained a conservative pellet enrichment tolerance of +0.05 wt% uranium-235.

g) The fuel assemblies contained no removable poisons. This assumption is required because ANSI/ANS-57.3-1983, Section 6.2.4.2 indicates that credit for poisons may only be taken for inherent structural materials or for poisons that cannot be accidentally removed by mechanical or chemical action. Since burnable poison clusters and control rods are not an integral part of the structural assembly design and can be mechanically removed, credit may not be taken for them.

h) Minimum assembly pitches between the two closest rows were assumed in the X and Y directions. The dimensions were reduced from nominal values to account for rack tolerance and off-center assembly placement.

i) No intermediate spacer grids were modeled. This is a conservative assumption since these materials contribute to parasitic neutron absorption.

j) No seismic events were considered since the NFSR is Class I seismically rated and changing the assembly enrichment does not change the racks' structural characteristics or tolerances.

4.2.2 NFSR Criticality Analysis Results

The criticality analysis results demonstrated that the new fuel storage area is critically safe for all normal and accident conditions with fuel enrichments up to and including 49.2 grams (4.75 w/o) of uranium-235 per axial centimeter. The limiting accident was found to be the new fuel storage area misted with 7% dense water concurrent with no assembly guides. The maximum k for that scenario, considering uncertainties and biases, was calculated to be 0.97666 with a 95/95 tolerance level. This is within the NRC imposed k-effective limit of 0.98 for misted conditions.

4.3 Criticality Analyses Conclusions

Criticality analyses were performed for both the SFSR and NFSR. These analyses demonstrated that the maximum fuel enrichment is limited by conditions in the NFSR. The limiting conditions were found to be the new fuel storage area misted with 7% dense water concurrent with no assembly guides. The resulting k-effective in the NFSR for those conditions was calculated to be 0.97666. By interpolation this would correspond to a "worst case" k-effective in the SFSR of 0.92339. These values are within the NRC imposed limits of 0.98 and 0.95, respectively. It should be noted that no administrative controls, such as burnup credit or checkerboarding, are necessary to ensure that these conditions are met.

5.0 Structural Considerations

The KNPP spent fuel and new fuel storage facilities, as well as the reactor vessel internals, are Seismic Category I, in accordance with the KNPP Updated Safety Analysis Report. Previous analyses have demonstrated that the fuel racks, when subjected to normal, abnormal and seismic loads within the KNPP design bases, will maintain their structural integrity. Increasing the fuel enrichment will not appreciably alter the mass of the fuel assemblies, and therefore will have no effect on the ability of the fuel racks to maintain the fuel assemblies in the proper configuration.

6.0 Spent Fuel Pool Thermal Evaluation

The Spent Fuel Pool Cooling System (SFPCS) was evaluated to assess its capabilities in cooling the pool with additional heat loads from higher enrichment fuel assembly core designs. The effects of the higher heat loads were evaluated in three areas: system performance, natural circulation cooling of the fuel assemblies and heating and ventilation system requirements.

6.1 SFPCS Analysis

The evaluation of the SFPCS was made assuming a worst case total core off-load to the spent fuel pool at the end of an equilibrium cycle. This maximized the decay heat output of the spent fuel. The SFPCS was evaluated under single failure conditions.

An acceptance criterion of a peak pool temperature less than or equal to 150°F was chosen as a conservative operating condition for the SFPCS. This value is based on component design temperatures of the SFPCS.

6.1.1 SFPCS Analysis Assumptions

The calculational models assumed the following conditions for the spent fuel pool cooling system analysis:

a) The spent fuel pool heat exchanger performance data was based on test data obtained at Kewaunee. This data allows credit for actual heat exchanger performance versus the design cooler performance. The effects of fouling on heat exchanger performance were inherent in the test data.

b) The KNPP service water was assumed to be 80°F based on the highest predicted service water system operating temperature. This is a conservative assumption since a high service water temperature results in a reduction in heat removal by the heat exchangers.

c) The decay heat load was based on the worst case design enrichment in the range up to 5.00 wt% U^{235} at a burnup of 52.5 Gwd/Mtu. During the analysis, it was found that the limiting case for decay heat load resulted from a 3.9 wt% U^{235} enrichment with a 52.5 Gwd/Mtu burnup. This was due to a combination of factors including the total number of fuel assemblies replaced each refueling, and the fuel assembly burnup. This enrichment maximized the heat load 100 hours after shutdown at the end of a cycle.

d) The decay heat load used was for a full core off-load. This places the largest demand on the spent fuel pool cooling system. Decay heat is dependent on enrichment and fuel burnup and not fuel assembly design.

6.1.2 SFPCS Performance Results

The SFPCS analysis showed that with a service water temperature of 80°F, sufficient heat removal capacity was available to maintain the spent fuel pool bulk liquid temperature below the system design temperature of 150°F. The spent fuel pool bulk liquid temperature was maintained at 141.2°F under conditions of a single failure in the SFPCS. This provides an 8.8°F margin to the system design temperature. The margin to the system design temperature increases with an increase in spent fuel pool heat exchanger flow rates.

6.2 Natural Circulation Analysis

To account for periods where failures within the SFPCS render the SFPCS unavailable, calculations to determine the heat removal from the fuel assemblies by natural circulation were made. The acceptance criteria for this evaluation were no bulk boiling and no localized fuel rod boiling. These limits are conservative for heat removal calculations, since adequate heat removal can exist with bulk boiling, however the acceptance criteria preclude the need to evaluate doses under vapor release conditions.

6.2.1 Natural Circulation Analysis Assumptions

a) The decay heat load used was based on worst case design enrichments up to 4.5 wt% U^{235} at a burnup of 52.5 Gwd/Mtu.

b) The decay heat load was multiplied by an enthalpy rise peaking factor of 1.52 to model an assembly in the high burnup region for determining the worst case fuel rack exit temperature.

c) The natural circulation path includes a heat sink to remove the decay heat load.

d) A nuclear peaking factor of 1.80 was applied to the decay heat load to model an assembly in a high burnup region for determining the margin to local fuel pin surface boiling.

6.2.2 Natural Circulation Results

The results of the natural circulation analysis indicate that the margin to bulk fluid boiling following a total loss of forced flow with a full core off-load is 34.9°F. The analysis indicates that a 5.3°F margin exists to local fuel rod surface boiling. The results show that sufficient natural circulation exists to remove decay heat for a full spent fuel pool containing a recent full core off-load, thus preventing local fuel rod surface and bulk liquid boiling.

6.3 Spent Fuel Building Heating and Ventilation Analysis

The heat load on the spent fuel building heating and air conditioning system was evaluated for equilibrium conditions with a single failure within the SFPCS. With a single train of the SFPCS unavailable, the average building temperature will be greater with higher burnup fuel. The evaluation conservatively predicted an average spent fuel building temperature.

6.3.1 Heating and Ventilation Analysis Assumptions

a) Heat transfer from the spent fuel pool to the building air was based on a natural convection heat transfer coefficient.

b) Spent fuel building ventilation air comes from the auxiliary building at a maximum temperature of 120°F.

c) Circulation flow rate was assumed to be 12,000 cfm.

d) Spent fuel building air volume was assumed to be 1.0×10^6 ft^3.

e) Air thermal properties used in the calculation were taken at 120°F and 14.7 psia.

6.3.2 Spent Fuel Building Air Temperature Results

The new decay heat loads resulting from an increase in the allowable fuel design enrichment to 4.5 wt% resulted in a higher spent fuel pool temperature under single failure conditions than was previously calculated. The higher pool temperature would result in a higher equilibrium spent fuel building temperature with the higher decay heat load.

The calculations showed that the spent fuel building equilibrium air temperature following a single failure of the spent fuel pool cooling system with a full pool and a recent full core off-load would be 121.7°F. This result places the building temperature 1.7°F above the nominal design condition. Since the design condition was used as an input temperature, this result is expected. The 1.7° higher temperature is not expected to adversely impact the operation of any equipment in the spent fuel building.

6.4 Spent Fuel Pool Thermal Evaluation Conclusions

The thermal evaluation of the spent fuel pool and building demonstrated the acceptability of increasing the fresh fuel design enrichment. The worst case heat loads for a full core off-load were predicted to occur with a 3.9 wt% U^{235} enrichment. The three areas examined were the SFPCS system performance, natural circulation cooling of the fuel assemblies, and spent fuel building temperatures. The worst case pool temperature with one SFPCS train inoperable was 141.2°F, 8.8°F below the acceptance criteria of 150°F, the limiting SFPCS component design temperature. Thus, during operation, the higher enrichment core designs will not lead to overheating of the spent fuel pool.

During natural circulation cooling of the fuel assemblies, with the SFPCS not operating, a margin of 34.9°F to bulk boiling was predicted. Additionally, the calculations predicted a 5.3°F margin to local fuel rod surface boiling. These results met the criteria for avoidance of bulk and surface boiling during natural circulation cooling.

Finally, the heat loads on the spent fuel building heating and ventilation system resulted in a worst case average building temperature of 121.7°F. Compared to the design temperature of 120°F assumed at the beginning of calculation, the spent fuel building temperature is not significantly affected by the failure of the ventilation system. This is due to the large volume of the building. Although a detailed review of the affected equipment was not made, this temperature increase is not expected to impact the operation of any of the equipment.

7.0 Radiological Analysis

Various radiological aspects of increasing the uranium enrichment of the nuclear fuel stored in the Kewaunee spent fuel pool (SFP) were evaluated. This section is divided into the following subsections:

1. Adequacy of Shielding
2. Evaluation of Fuel Handling Accident Doses
3. Adequacy of Spent Fuel Pool Ventilation System
4. Adequacy of Spent Fuel Pool Cleanup System

Increasing the uranium enrichment of the fuel has basically two effects that influence the radiological analyses: (1) the burnup dependent uranium/plutonium fission ratio changes. Since the fission product yields of plutonium and uranium are different, a change in the fission ratio results in a different total mixture of fission products with burnup, and (2) it enables the fuel to go to higher burnups which produces a larger inventory of long half-life nuclides.

Although higher burnup fuel can result in higher inventories of long half-life fission products, the fuel does not present a greater radiological risk. Three factors tend to mitigate the radiological risks associated with high burnup fuel: (1) the specific power and the linear heat rate decrease as the burnup increases (this significantly reduces the inventory of the short half-life nuclides that dominate the radiological dose consequences), (2) the neutron flux is lower due to the higher enrichment required for higher burnups, and (3) fewer high burnup fuel assemblies are required to generate the same amount of energy (this significantly reduces the effect of the larger inventory of long half-life nuclides). Typically increasing fuel burnup lowers the resulting radiological sources and dose rates.

7.1 Adequacy of Shielding

Shielding calculations were performed to verify the adequacy of the shielding provided by the concrete walls of the spent fuel pool, the 25 feet of water shielding the spent fuel in the storage racks, and the 10 feet of water shielding a single fuel assembly has while it is being moved in the spent fuel pool.

7.1.1 Shielding Sources

The gamma sources for a fuel assembly containing 18.953 Kg of U^{235} and 360.107 Kg of U^{238} (which would be typical of a KNPP fuel assembly with a 5 wt% U^{235} enrichment) were calculated. The calculations were based on a conservative power history and on a realistic power history. It should be noted that the very conservative power history imposes 6 weeks of operation at 20.8 MWt prior to each refueling outage (an average fuel assembly normally produces only 13.6 MWt).

For the conservative power history the gamma source strength increases with increasing burnup; whereas for the realistic power history the maximum gamma source strength occurs at the lowest burnup (i.e., at the highest linear heat rate).

In accordance with the Kewaunee Nuclear Power Plant Technical Specifications, fuel would have at least 100 hours of decay prior to any fuel handling operation. Additional gamma sources were developed using the conservative power history and a decay time of 100 hours. These gamma sources were evaluated to determine the burnup at which the maximum dose rate occurs.

7.1.2 Shielding by Concrete Walls

The gamma sources were used to calculate the gamma dose rate outside the concrete walls of the spent fuel pool. The resulting dose rates were for an infinite slab geometry, which is equivalent to a pool filled with fuel assemblies that were operating at 20.8 MWt (1.53 times the power of an average fuel assembly) and decayed for only 100 hours.

The gamma source that yielded the highest dose rate corresponded to a burnup of 48,000 MWD/MTU; however, the dose rate variation between the maximum and minimum values of burnup was only 3 or 4 percent. The highest calculated dose rate outside the spent fuel pool wall was 0.028 mr/hr.

7.1.3 Shielding by Pool Water

Calculations were also performed for a burnup of 48,000 MWD/MTU to determine the maximum dose rate at the surface of the spent fuel pool. The fuel pool was modeled as an infinite slab which corresponds to a pool filled with fuel assemblies that were operating at 20.8 MWt and decayed for 100 hours.

The dose rate at the surface of the water for those conditions was calculated to be 8.7×10^{-10} R/hr (which for all practical purposes is zero). The actual dose rate at the surface of the water is therefore attributable to radioactive material suspended or dissolved in the pool water.

For the case where a single fuel assembly is raised for movement to another location, the amount of water shield could be as little as 10 feet. The same gamma source that was used in the previous calculation was also used here (no credit was taken for the axial power shape which would reduce the source at the top of the fuel assembly). The dose rate at the surface of the water directly above the fuel assembly would be 10.2 mr/hr with 10 feet of water shielding. As the amount of water shielding is reduced the dose rate at the water surface increases by a factor of about 5.5 per foot of water.

7.1.4 Conclusions of Shielding Analysis

The use of fuel with an enrichment of 5 wt% U^{235} and with burnups of 48,000 or 60,000 MWD/MTU will not significantly increase any gamma dose rates and could result in lower dose rates. The analysis demonstrated that for conservative assumptions where high burnup and low burnup fuel operate at the same power, the radiation doses are within ±2 percent. If the high burnup fuel assemblies are assigned realistic values of power, dose rates would decrease as burnup increases.

7.2 Evaluation of Fuel Handling Accident

7.2.1 Accident Sources

The volatile radionuclides that would be released if the fuel rod cladding was ruptured are important to the analysis of the radiological dose consequences of a fuel handling accident. This source of volatile radionuclides is commonly referred to as the gap activity.

The gap activity used in this analysis was determined using two different methods. The first method used the methodology of Regulatory Guide 1.25. The Regulatory Guide defines the gap activity as 10% of the activity of each krypton, xenon, and iodine nuclide in the fuel, except for Kr^{85} where the value is 30% of its activity in the fuel. Using conservative and realistic power histories calculations were performed for 5 wt% U^{235} enriched fuel to determine the activity of the krypton, xenon, and iodine nuclides in the fuel. The gap activities were then determined using the specified percentage per Regulatory Guide 1.25.

The second method was based on ANSI/ANS-5.4-1982. This ANSI Standard correlates the release of activity into the gap as a function of time, temperature, and burnup. Conservative and realistic power histories were used to define the time, burnup, and linear heat rate parameters used to calculate the axial and radial temperature distribution in the fuel rods. These temperature distributions were then used to calculate the gap activities.

High power densities lead to very high fuel temperatures which in turn lead to very high release rates into the gap. At realistic power densities, the temperatures are much lower, implying that the ANSI standard predicts much lower gap activities.

7.2.2 Accident Doses

The doses at the Exclusion Area boundary were calculated for each of the source terms.

The two-hour thyroid dose at the Exclusion Area boundary was calculated as a function of burnup for the source terms.
The thyroid dose for each source agreed reasonably well at low burnups and diverged as the burnup increased. This divergence was a direct result of using high, unrealistic power densities at high burnups. Since the Regulatory Guide 1.25 methodology is independent of fuel temperatures, high power densities

and/or high linear rates only increase the fission product generation rate in the fuel but do not increase the fraction that enters the gap region. For the conservative case, the linear heat rate is assumed to be 9.7 Kw/ft for 6 weeks prior to shutdown at all burnups. The increase in the fission product source is the result of a slight change in the total fission yields due to the shift to a higher plutonium to uranium fission ratio. The doses calculated using the Regulatory Guide 1.25 approach are therefore relatively constant with respect to burnup. For realistic power histories where the linear heat rate decreases from 9.7 to 4.0 Kw/ft as the burnup increases, the Regulatory Guide methodology yields doses that decrease with burnup. The decrease is almost directly proportional to the power reduction except for the small perturbation caused by the differences in the fission yields of plutonium and uranium.

In the methodology defined in the ANSI Standard, the fission product release rate into the fuel rod gap is a very strong function of temperature and burnup. Therefore, at high linear heat rates, the temperature is high and the release rate into the gap is high. As the burnup increases, the release rate into the gap would tend to increase exponentially except the increase is suppressed by the decreasing linear heat rate. The decreasing linear heat rate decreases the fuel temperatures and causes the release rate into the gap to actually decrease. For the conservative case where the linear heat rate is constant 9.7 Kw/ft for 6 weeks prior to shutdown at all burnups, the ANSI methodology shows that the fission product release rate increases rapidly with burnup, thus the doses increase by a factor of 10 in going from 14,000 to 60,000 MWD/MTU. For the realistic case where the linear heat rate decreases as the fuel becomes expended at higher burnups, the resulting decrease in temperature reduces the fission product release rate into the gap to a much greater extent than the effects of burnup can increase the release rate. Thus, as the linear heat rate decreases from 9.7 Kw/ft at 14,000 MWD/MTU to 6.8 Kw/ft at 48,000 MWD/MTU, the doses decrease by approximately a factor of 4. However, as the linear heat rate decreases from 6.8 Kw/ft at 48,000 MWD/MTU to 4.0 KW/ft at 60,000 MWD/MTU, the temperatures become so low that the fission product release rate plummets, and the doses fall by more than a factor of 50.

Of the cases analyzed, the soundest technical basis is the ANSI methodology with realistic power histories. The ANSI methodology with conservative power histories shows that the assumption of operating at 9.7 Kw/ft for six weeks prior to each shutdown is too conservative to be credible. The Regulatory Guide methodology does not accurately reflect the effects of increased burnup, nevertheless the doses calculated using the Regulatory Guide methodology have been the industry standard for licensing.

Finally, it should be noted that, regardless of method used to calculate the sources, all of the doses are well below the acceptance criteria of 75 rem to the thyroid and 6 rem to the whole body stated in Section 15.7.4 of the NRC's Standard Review Plan.

7.3 Adequacy of Spent Fuel Pool Cleanup System

The adequacy of the spent fuel pool cleanup system had been previously evaluated. That evaluation was based on the pool capacity of 990 spent fuel assemblies and an enrichment of 3.5 wt% U^{235}. The analysis performed to support the evaluation was very conservative because it calculated activity build up over 22 cycles using a SFP demineralizer removal efficiency of zero for Co^{58} and Co^{60} (which constitute about 87% of the dose rate), a removal efficiency of only 50% for Cs^{134} and Cs^{137} (which constitute about 12% of the dose rate), and a removal efficiency of 90% for all other nuclides. This conservative analysis yielded a maximum dose rate of 112 mr/hr at five feet above the surface of the pool.

The conservative dose evaluation described above was based on a total activity of 0.10 microcuries per milliliter in the spent fuel pool water. Twenty-six measurements of the total activity in the water of the spent fuel pool, which were taken over a 3 year period, showed a mean value of 0.0073 microcuries per milliliter (with a standard deviation of 0.0057 microcuries per milliliter). Thus, the actual dose rate at the surface of the pool would more typically be in the 2 to 15 mr/hr range. These lower dose rates are also supported by radiation dose rate measurements.

The most likely reason for the actual dose rates being much lower than the dose rates previously calculated is that the SFP demineralizers actually provide a much higher removal efficiency than the calculations assumed. Measurements of the SFP demineralizer performance show that they can provide an overall removal efficiency of greater than 99%. Thus, the assumption that the removal efficiency was zero for the cobalt nuclides was much too conservative.

The general effect of increasing the uranium enrichment to 5 wt% U^{235} was evaluated relative to the results for 3.5 wt% U^{235}. The use of fuel with a higher U^{235} enrichment will reduce the release rate of cobalt and other radionuclides into the pool water, thus it will tend to result in a slight reduction in the dose rate at the surface of the pool and the SFP demineralizers. This reduction is primarily due to the fact that fuel at a higher enrichment operates with a lower neutron flux when generating the same power as fuel with a lower enrichment. A lower neutron flux reduces the generation rate of all radionuclides produced by neutron activation reactions; thus smaller amounts of Co^{58} and Co^{60} activity will be generated. Since the currently planned increase in enrichment is only from 3.5 wt% to 4.2

wt% U^{235} (not from 3 wt% to 5 wt% U^{235}), a dose rate reduction of only about 7% would be expected due to the lower neutron flux. Higher enrichment levels will reduce the number of fuel assemblies that are replaced each cycle. With fewer assemblies added to the spent fuel pool, it might be expected that the associated cobalt levels added each cycle would also be reduced. However, any decrease due to this would be offset by an increase in the build up of the cobalt activity due to the longer irradiation time.

The activity of short half-life fission products is determined primarily by the power (i.e., fission rate) generated by the fuel assembly and to a small extent by the total fission yield of various uranium and plutonium nuclides. In general, the spent fuel with higher burnup will have been operating at a relatively low power immediately prior to discharge. Thus, the amount of short half-life activity in the SFP will be much lower because there will be fewer fuel assemblies discharged each cycle and each will contain a smaller inventory of short half-life radionuclides.

The activity of long half-life fission products (i.e., Cs^{137}) is directly proportional to the total number of fissions that have occurred in the fuel assembly (i.e., proportional to the burnup and/or the amount of energy produced by the assembly). Since each high burnup (i.e., high enrichment) fuel assembly will produce more energy, it will have a proportionately greater Cs^{137} inventory; however, this increase in inventory will be offset by the smaller batch size. Thus, the only effect that going to a higher burnup will have is that the smaller batches will enable the pool to accommodate two more cycles of operation before it is full. In essence, this will increase the total Cs^{137} inventory in the pool by about 5% when the pool is completely full. Since the Cs^{137} accounts for only about 7% of the dose rate, the net effect of the 5% higher inventory is less than 0.5% increase in total dose rate. This small increase in dose rate is negligible since it is overshadowed by other controllable factors (i.e., cleanup flow rate and/or removal efficiency).

In conclusion, the spent fuel pool cleanup system can control the radionuclide activity in the pool water at or below current levels with higher fuel enrichment.

7.4 Adequacy of Spent Fuel Pool Ventilation System

The impact of increasing the fuel enrichment to 5 wt% U^{235} and the average burnup to 48,000 MWD/MTU on the adequacy of the spent fuel pool (SFP) ventilation system was also evaluated.

The SFP ventilation system is designed to limit the exposure of plant personnel to radioactive gases and/or aerosol particles that may escape from spent fuel stored in the SFP primarily during fuel movement. This is accomplished by having an air supply fan blow air across the pool toward exhaust registers located near the surface of the pool. Radioactive materials that might be released from the spent fuel will be drawn into the exhaust registers and through HEPA (high efficiency particulate air) filters and charcoal filters by two exhaust fans. The dual train exhaust system will remove greater than 99% of the radioactive aerosol particles and greater than 90% airborne iodine activity prior to discharging the exhaust air to the atmosphere via the Auxiliary Building vent.

The performance of the SFP ventilation system has two separate and distinct aspects: (1) With regard to the protection of plant personnel in the SFP area, the performance of the SFP ventilation system is almost solely dependent on the efficiency of the curtain sweep air for trapping gases emanating from the surface of the pool. The performance is essentially independent of the flow rate (other than the flow rate required to maintain an effective air curtain) and independent of the efficiency of the HEPA and charcoal filters. (2) With regard to the protection of persons offsite, the performance of the SFP ventilation system is dependent on the efficiency of the HEPA and charcoal filters.

Increasing the U^{235} enrichment of the fuel and increasing the burnup of the fuel does not necessarily result in a larger fission product source or in higher dose rates. The effect of increasing the enrichment of U^{235} in the fuel can be seen by comparing the activity inventory in two fuel assemblies that have different enrichments but have identical power histories and burnups. The activity inventory of the significant gaseous or volatile fission product nuclides (after 3 days of decay) for a fuel assembly with an enrichment of 3 wt% U^{235}

and for a fuel assembly with an enrichment of 5 wt% U^{235} were calculated. The most significant effects of this enrichment increase can be summarized as follows:

(1) the Kr^{85} increases by 7% - 18% as the burnup increases from 8,000 to 60,000 MWD/MTU;

(2) the I^{131} decreases 3-4% at all burnups;

(3) Xe^{131m} and I^{132} decrease 3-5%, and Xe^{135} increases by 1-3%, (these three nuclides have essentially no dose impact); and

(4) the other nuclides change by less than $\pm$1% and will have no significant dose impact.

The conclusions to be drawn from this comparison of the effects of increasing the enrichment are that:

(1) If a fission product release were to occur 3 days after shutdown, the thyroid doses in the SFP area and offsite would be 3-4% lower for the fuel with the higher enrichment, and the whole body doses would be within $\pm$1% (because the Kr^{85} has a negligible dose contribution relative to the other short half-life nuclides).

(2) If a fission product release were to occur after several months of decay, the decay of the iodine activity would make the thyroid doses negligible for both enrichments, and although the whole body dose would be very low, it would be 7% to 18% higher for the fuel with the higher enrichment due to the greater amount of Kr^{85}.

Based on this evaluation, it seems clear that increasing the enrichment and burnup of the fuel will result in either an insignificant increase or a decrease in the onsite and offsite doses depending upon the circumstances. For example, a fuel assembly with a 5 wt% U^{235} enrichment and a burnup of 48,000 MWD/MTU would have about 37% more Kr^{85} than an assembly at 3 wt% U^{235} enrichment and a burnup of 38,000 MWD/MTU. Thus, for a situation where a 48,000 MWD/MTU fuel rod with a 5 wt% U^{235} enrichment failed after several months of decay in the SFP, the whole body dose could be 37% higher than for a similar failure of a 38,000 MWD/MTU fuel rod with a 3 wt% U^{235} enrichment, but the whole body dose from the failure of either of these fuel rods would be very low. If the same fuel rods were to fail with only a few days of decay, the whole body doses from the two fuel rods would be much higher, but both would be almost identical, because Kr^{85} contributes very little to the total whole body dose due to its low dose conversion factor and its small activity inventory.

In conclusion, the current SFP ventilation is adequate for use with fuel assemblies containing UO_2 enriched to 5 wt% U^{235} and irradiated to between 48,000 and 60,000 MWD/MTU. This conclusion is based primarily on the fact that the most significant difference is the amount of Kr^{85}, and Kr^{85} does not present a significant whole body dose concern when compared to the whole body doses that would result from the radionuclide inventory in the current fuel at short decay times.

7.5 Conclusions of Radiological Analysis

Based on the radiological analyses performed for the Kewaunee spent fuel pool, the following conclusions can be drawn relating to the storage of fuel with U^{235} enrichments up to 5 wt% and with burnups up to 60,000 MWD/MTU.

1. Higher enrichments and higher burnups do not significantly increase the radiation source or the radionuclide inventory in spent fuel assemblies.

2. The dose rates outside of the shielding walls and above the surface of the water in the pool should not change significantly.

3. The offsite dose rates are greatly dependent on the conservatism in the assumptions, but even with the most conservative assumptions all doses are well below the acceptance criteria in Section 15.7.5 of NRC's Standard Review Plan.

4. The SFP cleanup system is adequate for controlling the activity in the pool water at or below current activity levels.

5. The SFP ventilation system is adequate to provide the current level of protection with high burnup fuel assemblies in the pool.

8.0 Significant Hazards Determination

Wisconsin Public Service Corporation has reviewed this proposed change in accordance with 10CFR50.92 and concluded that it does not involve a significant hazards consideration in that this change would not:

a) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The criticality analysis which was performed in support of this proposed change demonstrated that adequate margins to criticality can be maintained with fuel enrichments up to 4.75 weight percent of uranium-235 stored in the new fuel and spent fuel storage racks.

The bounding cases of the analyses demonstrated that Keff remains less than 0.95 in the spent fuel pool and less than 0.98 in the new fuel storage pit. Therefore, the 4.75 weight percent enrichment is acceptable.

Other than criticality, the only other accident that need be considered is a fuel handling accident. Since the mass of the fuel assembly would not be appreciably altered by the increased fuel enrichment, the probability of this accident occurring is not changed. Because fission product inventories in a fuel assembly are not a significant function of initial fuel enrichment, the consequences of a fuel handling accident also would not be affected by the use of higher fuel enrichment.

It should be noted here that any changes in the nuclear properties of the reactor core that may result from higher fuel enrichments would be analyzed in the appropriate reload analysis.

b) Create the possibility of a new or different kind of accident from any accident previously evaluated.

As discussed above, the only safety issue significantly affected by the proposed change is the criticality analysis of the spent fuel storage pool and new fuel storage pit. Since it has been demonstrated that Keff remains below 0.95 and 0.98 respectively in those areas, no new or different accident would be created through the use of fuel enrichments up to 4.75 weight percent uranium-235 at the Kewaunee Nuclear Power Plant.

c) Involve a significant reduction in a margin of safety.

Since the analyses have shown that increasing the allowable weight percent enrichment to 4.75 would not increase Keff above 0.95 in the spent fuel storage pool and 0.98 in the new fuel storage pit, it is concluded that this proposed change would have no impact on the margin of safety as defined in the basis for any Kewaunee Nuclear Power Plant Technical Specification. Any changes in the nuclear properties of the reactor core that may result from higher fuel enrichments would be analyzed in the appropriate reload analysis to ensure compliance with applicable reload considerations and requirements.

In conclusion, the analyses performed in support of the proposed change have demonstrated that increasing the maximum allowable fuel enrichment at the Kewaunee Nuclear Power Plant to 49.2 grams (4.75 weight percent) of uranium-235 per axial centimeter of fuel assembly does not involve a significant hazards consideration.