UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE MODIFICATION OF THE SPENT FUEL POOL

FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

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1.0 Introduction

By letter dated March 12, 1982, as supplemented by letters dated April 26, 1982, June 23, 1982, November 23, 1982, January 21, 1983, April 29, 1983, June 6, 1983, August 1, 1983 and August 23, 1983, the Omaha Public Power District (the licensee) proposed changes in the spent fuel storage capacity for the Fort Calhoun Station, Unit No. 1. The changes consist of replacing the existing storage racks with new storage racks (reracking).

Fort Calhoun currently has the capacity for storing 483 fuel assemblies. This represents an increase from the 178 assembly capacity, approved when Fort Calhoun was licensed, which was accomplished by reracking the spent fuel pool as approved by the NRC in Amendment No. 13 to the operating license. With the existing storage capacity, the capability to discharge the entire reactor core into the spent fuel pool would be lost in 1985. However, from a logistics point of view (removing the current racks, inserting the new racks, considering that there are 265 spent fuel assemblies currently being stored, and also maintaining a full core discharge capability), the licensee must complete the installation before the 1984 outage, which is estimated to commence in early 1984. Due to these concerns and the present lack of commercial reprocessing or available commercial central storage facilities for spent nuclear fuel, the licensee proposes to increase the spent fuel pool capacity of the Station to 729 fuel assemblies. This would be accomplished by replacing the existing racks with higher density stainless steel racks which will utilize a neutron poison material in some of the racks for reactivity control. The use of the new racks will extend the spent fuel storage and full core discharge capability of the Station to the year 1994. The new rack design would also permit the implementation of disassembly and compact storage of individual rods in canisters beginning in 1994. This would accommodate all spent fuel through the year 2008, which is the year that the current license expires. Although the design of the racks permits this type of storage, the licensee is proposing at this time only to expand the storage capacity to 729 whole spent fuel assemblies.

The following safety evaluation addresses the licensee's proposal. A separate environmental impact appraisal addressing these changes has also been prepared.

2.0 Evaluation

2.1 Criticality Considerations

The spent fuel storage racks are divided into two regions. Region 1 utilizes the neutron absorber, Boraflex, and can accommodate the storage of any Fort Calhoun fuel assembly having an enrichment of 4.0 weight percent U-235

or less. It is intended to be used for a full core discharge and for failed or damaged fuel. This region could also be used for temporary storage of freshly discharged fuel. Region 2 does not contain any neutron poison material and storage in this region is restricted by burnup and enrichment limits. Placement of fuel in Region 2 will be determined by burnup calculations and controlled administratively by the licensee. The new racks would expand the pool storage capacity to 729 fuel assemblies as compared to the current capacity of 483 assemblies. The criticality aspects of the design of each region are discussed separately below.

2.1.1 Region 1 Design

The proposed Region 1 racks consist of individual stainless steel storage cells with a neutron absorbing material, Boraflex, attached to each cell. The Boraflex contains a minimum of 0.020 grams of B-10 per square centimeter of surface area. There are 198 fuel assembly storage locations with a nominal 9.935 inch center-to-center spacing between assemblies. For the criticality analysis, the racks are assumed to be filled with unirradiated Combustion Engineering (CE) 14X14 fuel assemblies having an enrichment of 4.0 weight percent U-235. Calculations are performed for unborated water in the pool which is at the lowest anticipated temperature of 68°F.

The reactivity calculations were performed with the PDQ-7 diffusion theory computer code with neutron cross sections generated by the LEOPARD code. These codes were verified by comparison with critical experiments. Experiments containing stainless steel and boral plates between assembly mockups with a range of U-235 enrichments and moderator-to-fuel ratios encompassing those in the Fort Calhoun storage racks were analyzed. In addition, more highly absorbing experiments containing absorber plates of cadmium, gold, and hafnium with reactivity worths comparable to the Boraflex worth in the Fort Calhoun spent fuel racks were analyzed. From these comparisons, calculational biases and uncertainties were obtained. The uncertainty correction based upon the sum of the bias and twice the standard deviation was nearly identical for both sets of experiments. However, since the calculational correction for the first set of experiments was found to be more conservative than that for the latter experiments, it was used to determine the final multiplication factor of the Fort Calhoun racks.

The effective multiplication factor for the nominal cell was calculated to be 0.9258. When the calculational bias is added, the result is 0.9319. When the effect of uncertainties in the box-to-box spacing, in stainless steel thickness and in fuel pellet density and diameter are combined with the calculational uncertainty and added to the biased value, the effective

multiplication factor, including all biases and uncertainties, is 0.9415. The uncertainties are taken at the 95 percent probability level with 95 percent confidence. This meets our acceptance criterion of less than or equal to 0.95 for this quantity and is, therefore, acceptable. We conclude that any number of fuel assemblies of the CE 14X14 design having enrichments no greater than 4.0 weight percent U-235 may be stored in Region 1 of the racks.

2.1.2 Region 2 Design

The proposed Region 2 racks consist of stainless steel storage cells. There are 531 fuel assembly storage locations with a nominal 9.935 inch center-to-center spacing between assemblies. The same methods were used for the basic reactivity determination as used in the Region 1 analysis. In addition, the LEOPARD/CINDER codes were used to calculate the isotopic composition and neutron cross sections of the fuel as a function of burnup history and subsequent decay time. Direct verification of the codes was not possible because no critical experiments have been done with assemblies having large burnups. Rather, verification of various aspects of the calculation was undertaken. For example, the ability to calculate the isotopic composition of irradiated fuel was verified by comparing LEOPARD/CINDER calculations to the measured results of irradiations performed on UO_2 fuel in the Yankee-Rowe reactor on mixed oxide (PuO_2-UO_2) fuel in the Saxton reactor. Criticality analyses for the mixed oxide Saxton and Esada measurements were used to assess the accuracy of reactivity calculations for irradiated fuel. Also, calculations of reactor reactivity lifetimes were used to estimate the uncertainty associated with the calculated reduction in fuel assembly reactivity with the depletion of the heavy metals and the accumulation of fission products as a function of fuel assembly exposure. Reactivity perturbations associated with manufacturing and thermal tolerances and uncertainties were re-evaluated for depleted fuel in the Region 2 configuration. These were based on a fuel cell of 4.0 weight percent enriched fuel at an average burnup of 36,000 MWD/MTU and at a pool temperature of 68°F. The total uncertainty was determined to be .0365 Δk.

In order to establish burnup criteria for storage in Region 2, the infinite multiplication factor (at a pool temperature of 68°F with minimum post-shutdown fission product inventory) of a fuel assembly as a function of burnup is obtained for a number of initial enrichments. The multiplication factor of the storage racks is next calculated as a function of the infinite multiplication factor of the stored assemblies for the various initial

enrichments. The assembly infinite multiplication factor which produces a rack multiplication factor of 0.91 is obtained from these curves as a function of initial enrichment. This rack multiplication factor is chosen so as to remain below our acceptance criterion of 0.95 when the above uncertainty of .0365 is added. Finally, a curve is drawn of the burnup required to produce a rack multiplication of 0.91 as a function of initial enrichment. This curve, shown in the attached Figure 2-10, is approximately linear and yields required burnups of 17,000 MWD/MTU for a 2.5 weight percent U-235 enrichment and 35,700 MWD/MTU for a 4.0 weight percent U-235 enrichment.

One dimensional axial calculations of a fuel assembly with axially dependent neutron cross sections and with an axially uniform set of neutron cross sections, both corresponding to an average burnup of 15,000 MWD/MTU, were performed. Based on these calculations, the use of a simple assembly average exposure resulted in an over-estimate of the fuel assembly effective multiplication constant by about +.015 $\Delta k/k$ compared to a more realistic non-uniform axial burnup calculation and is, therefore, conservative.

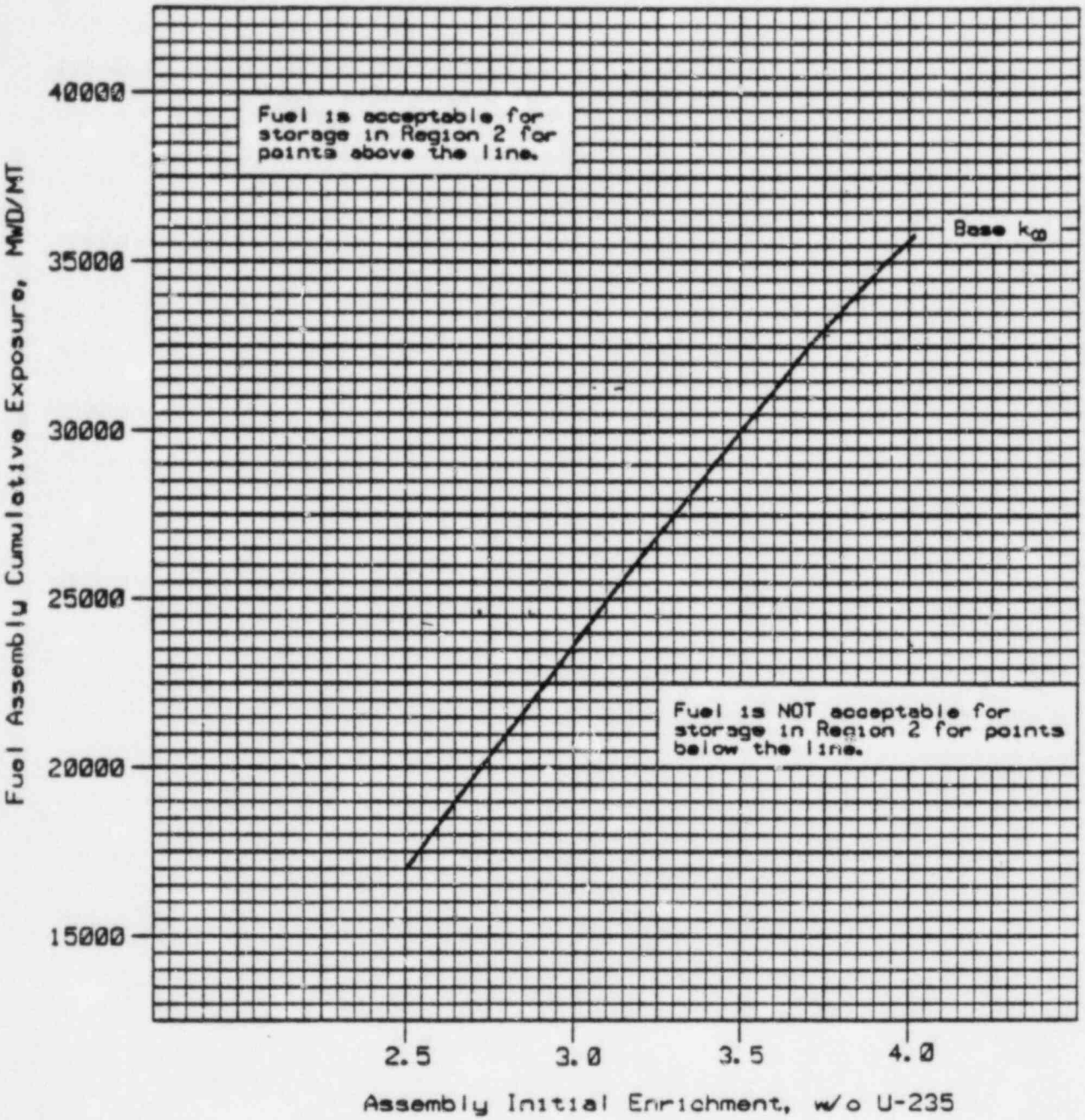
Based on our review, we conclude that any number of CE 14X14 designed fuel assemblies with burnups in the acceptable region of the attached Figure 2-10 may be stored in Region 2 of the Fort Calhoun spent fuel racks.

2.1.3 Criticality Accident Anlysis

The effect of accidental placement of fuel assemblies in other than their intended location has been considered. Dropping a fuel bundle on top of the racks will have a negligible effect on the effective multiplication factor since the fuel in the stored assemblies is more than two feet below the tops of the racks. Stand-offs have been provided to prevent placement of an assembly adjacent to the racks.

Loss of all cooling systems would result in a temperature increase of the pool water and possible boiling. Because of the heavy boron loading in Region 1, temperatures higher than the base case value of 68°F cause a reduction in the multiplication factor. Likewise, the full water density assumed in the base case calculation produces the most reactive condition. For Region 2, an additional bias was included to account for the most reactive temperature condition over the possible operating range. Therefore, a loss of all cooling systems would not produce a criticality accident.

Spent Fuel Pool Region 2 Storage Criteria
Minimum Required Fuel Assembly Exposure
as a Function of Initial Enrichment
to Permit Storage in Region 2



SPENT FUEL POOL REGION 2 STORAGE CRITERIA	OMAHA PUBLIC POWER DISTRICT FORT CALHOUN STATION-UNIT No. 1	FIGURE 2-10

2.1.4 Administrative Control of Storage in Region 2 and Proposed TSs

The decision as the whether a particular fuel assembly is to be placed in Region 2 of the spent fuel storage racks is proposed to be made under administrative control. After the assemblies to be discharged have been transferred to Region 1 of the racks from the core, an analysis of the burnup of each assembly will be made. This analysis will make use of the records which show the location of each assembly at all times since its arrival on site and of core operating histories and power distributions while the assembly was in the reactor. The records of fuel assembly location are generated by following written, previously approved procedures. The burnup history of each assembly is integrated in order to arrive at its cumulative exposure (MWD/MTU). The burnup value is then compared to the storage acceptance criterion (Figure 2-10) in order to permit or deny storage in Region 2. This figure will be included in the Fort Calhoun Technical Specifications. No freshly discharged fuel will be placed in Region 2 until minimum assembly exposure has been verified by calculation. To prevent freshly discharged fuel from being placed in Region 2, the fuel handling machine will be interlocked to prevent movement to that region during refueling operations.

The above requirements and procedures have been satisfactorily included in the Fort Calhoun proposed Technical Specifications. We, therefore, find the use of administrative control to select assemblies for storage in Region 2 of the spent fuel racks to be acceptable.

2.1.5 Conclusions

We conclude that the proposed storage racks meet the requirements of General Design Criterion 62 as regards criticality. This conclusion is based on the following considerations:

1. State-of-the-art calculation methods which have been verified by comparison with experiment have been used.

2. Conservative assumptions have been made about the enrichment of the fuel to be stored and the pool conditions.

3. Credible accidents have been considered.

4. Suitable uncertainties have been considered in arriving at the final value of the multiplication factor.

5. The final effective multiplication factor value meets our acceptance criterion.

2.2 Spent Fuel Pool Cooling and Makeup

2.2.1 Introduction

The spent fuel contained in the spent fuel pool is cooled by the spent fuel pool cooling and cleaning system. The system consists of two storage pool circulation pumps, a storage pool heat exchanger, a demineralizer, a filter, two fuel transfer canal drain pumps, piping, valves, and instrumentation. The pumps circulate the pool water through the heat exchanger and return it to the pool. The fuel transfer canal drain pumps can be used to provide make-up water to the spent fuel pool from the safety injection and refueling water tank. This section provides our evaluation for all components used to cool all 729 fuel assemblies except for the cleanup part of the system, which is addressed in section 2.6.

2.2.2 Evaluation

The licensee's calculated spent fuel discharge heat load to the pool, which was determined in accordance with the Branch Technical Position ASB 9-2 "Residual Decay Energy for Light Water Reactors for Long Term Cooling," and the Standard Review Plan Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," indicates that the expected maximum normal heat load following the last refueling is 11 MBtu/hr. This heat load results in a maximum bulk pool temperature of 120°F. The expected maximum abnormal heat load following a full core discharge is 22 MBtu/hr. This abnormal heat load results in a maximum bulk pool temperature of 150°F. When a full core is discharged to the spent fuel pool, the Shutdown Cooling System (SCS) can provide supplemental cooling in addition to the Spent Fuel Pool Cooling System and thereby maintain the pool temperature below 150°F. The SCS system will be initiated upon high pool water temperature alarm (150°F). Assuming a pool temperature of 150°F and two hours to start SCS, the maximum pool temperature will be 175°F.

The spent fuel pool water is cooled by the component cooling water system, which in turn is cooled by the raw water system. The licensee proposed no modifications to these two systems as part of this spent fuel pool

expansion. These systems were reviewed as to their adequacy to remove the additional heat load and were found to be capable of removing the additional heat.

2.2.3 Conclusion

Based on the above, we conclude that the proposed spent fuel pool modification is acceptable for storage of 729 spent fuel assemblies with respect to the storage rack capacity, the developed heat decay loads, and the spent fuel pool cooling and support system capabilities.

2.3 Installation of Racks and Load Handling

2.3.1 Description

The proposed spent fuel storage modification will provide storage locations for 729 fuel assemblies. Region 1 is designed to accommodate fresh fuel, damaged fuel, and the reactor fuel if the reactor pressure vessel has to be unloaded. Storage in Region 2 is restricted by burnup and enrichment limits. There will be no physical barrier between the two regions. The storage racks will have three basic sizes (8X9, 7X9, and 6X9). The 8X9 rack will contain 72 cells (8X9), weigh 15,800 lbs., and have the planar dimensions of 7 1/2 by 6.7 feet. There will be only one 8X9 rack; it will contain Boraflex and will be located in Region 1. The 7X9 rack will contain 63 cells (7X9), weigh 13,825 lbs., and have the planar dimensions of 7 1/2 by 5.9 feet. There will be seven 7X9 racks; two will contain Boraflex and will be located in Region 1. The other five will not contain Boraflex and they will be located in Region 2. The 6X9 rack will contain 54 cells (6X9), weigh 11,850 lbs., and have the planar dimensions of 7 1/2 by 5 feet. There will be four 6X9 racks and they will not contain Boraflex. These four racks will be located in Region 2. The cell pitch for each rack is 9.935 inches and prevents placement of a fuel assembly anywhere other than a design location. One whole fuel assembly will be inserted in each cell.

2.3.2 Rack Handling and Installation

The review of heavy load handling at Fort Calhoun is being conducted as part of the ongoing generic review initiated by NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The results of that review will be reported as part of Multiplant Action Item C-10. The evaluation below is limited to the heavy load handling activities associated with the proposed spent fuel pool modifications.

Fort Calhoun has one seismic Category I overhead crane in the auxiliary building which will be used for removing the existing racks and lowering the new racks into the pool. All movement of spent fuel racks will be controlled by written administrative procedures which will prohibit movement of the racks over locations in the pool where fuel is stored. The licensee has provided for our review a procedure for safe load paths for removal and installation of spent fuel pool racks. We find that the re-rack procedures and heavy load paths are acceptable.

2.3.3 Conclusion

We have reviewed the described load handling operations and equipment needed for the rack modifications. As part of this review, we have reviewed a procedure on safe load paths submitted by the licensee. We conclude that the re-rack procedure and heavy load paths are acceptable.

2.4 Structural Design

2.4.1 Introduction

The Fort Calhoun reactor is a PWR. The spent fuel pool is located in the auxiliary building just outside the containment. The pool is a reinforced concrete structure which is 20'-7" wide by 33'-3" long by 43'-0" deep. Wall thicknesses are 5'-6" on 3 sides and 4'-0" on one side. The floor is 12'-0" thick and is supported by steel piles which rest on sound rock.

The pool is lined with a welded stainless steel watertight liner plate which varies in thickness from 1/16 inch to 3/32 inches. The liner plate is backed by, welded to a grid of stainless steel angles and plates. A leak-chase system is provided behind the liner in order to detect leaks.

Existing spent fuel racks are to be removed from the spent fuel pool and replaced with either poisoned or unpoisoned free-standing racks. The pool is to be divided into two regions. Region one consists of 3 poisoned racks designed to store 198 fuel assemblies. Region two consists of 9 unpoisoned racks with a capacity of 531 fuel cells. Thus the total pool capacity will be 729 fuel assemblies. The current pool capacity at Fort Calhoun is 483 fuel assemblies.

The new racks are stainless steel "egg-crate" structures. The 7 cell by 9 cell rack is about 14.2 feet high by about 7.5 feet long by about 5.9 feet wide. The cells of the egg-crate are fabricated of cold-formed gage thickness material. These cells are supported on a heavy welded base. A heavy flat-bar extends around the perimeter of each rack near the top. The racks are each free-standing on the pool floor on four corner leveling pads.

2.4.2 Applicable Codes, Standards and Specifications

Structural material of the racks were compared with the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF (hereafter referred to as the "ASME Code"). Load combinations and acceptance criteria were compared with those found in the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and amended January 18, 1979 (hereafter referred to as the "NRC Position"). The existing concrete pool structure was evaluated for the new loads in accordance with the requirements of the American Concrete Institute Code, ACI 318-71.

2.4.3 Loads and Load Combinations

Loads and load combinations for the racks and the pool structure were reviewed and found to be in agreement with the applicable portions of the NRC Position.

2.4.4 Seismic and Impact Loads

Seismic loads for the rack design are based on the original design floor acceleration response spectra calculated for the plant at the licensing stage. This was based on a 0.17g SSE and a 0.08g OBE. Damping values for the seismic analysis of the racks and the pool structure were taken as 2 percent for OBE and 4 percent for SSE. Rack/fuel bundle interactions were considered in the structural analysis.

Loads due to a fuel bundle drop accident were considered in a separate analysis for such an occurrence.

The postulated loads from such events were found to be acceptable.

2.4.5 Design and Analysis Procedures

a. Design and Analysis of the Racks

First, a simplified 2 dimensional analysis of the racks was conducted in order to define worst-case loading conditions for eccentric fuel loading in the racks and varying coefficients of friction. Next, a non-linear 3 dimensional time-history analysis of a two rack model was accomplished. The model included mass, spring, damping, and gap elements and accounts for sliding, tipping and potential rack-to-rack interaction. A detailed finite-element model of the racks was also constructed and a linear time-history analysis was performed for the imposed loads in order to determine stresses and strains within the racks.

Calculated stresses for the rack components were found to be well within allowable limits. The racks were found to have adequate margins against sliding and tipping.

An analysis was conducted to assess the potential effects of a dropped fuel bundle on the racks and results were considered satisfactory.

An analysis was conducted to assess the potential effects of a stuck fuel assembly causing an uplift load on the racks and a corresponding downward load on the lifting device as well as a tension in the fuel bundles. Resulting stresses were found to be within acceptance limits.

b. Analysis of the Pool Structure

The licensee did not indicate that it performed a re-analysis of the spent fuel pool structure for the new rack loads. However, an assessment of support pile loadings for a loading condition involving implementation of a "disassembly and compacted storage" (DCS) scheme showed that the pile loadings would be within allowable load limits. The staff's understanding is that a DCS scheme is not postulated at this time and thus only the lesser loads of high-density racks will be imposed on the pool floor. Given that the staff's evaluation considers only installation of high-density storage racks without DCS, the increase in pool structure loading is a small percentage of the weight of the pool and water weight. Also, the staff's observation of the pool structure design indicates that the pool floor and walls are conservatively designed. Accordingly, for the high-density rack installation with standard spent fuel bundles, which the licensee now proposes, the staff considers the pool structure to be acceptable without additional re-analysis. If, in the future, the

licensee proposes to implement a DCS scheme, the staff will then require a full re-analysis of the loads and stresses in both the pool structure and racks. This should include a seismic reanalysis of the loads and stresses in both the pool structure and racks in order to assure that adequate margins of safety are maintained.

2.4.6 Conclusion

It is concluded that the proposed rack installation will satisfy the requirements of 10 CFR 50, Appendix A, GDC 2, 4, 61 and 62 as applicable to structures, and is therefore acceptable.

2.5 Materials

2.5.1 Materials Description

The proposed spent fuel storage racks are fabricated of Type 304 stainless steel, which is used for all structural components. The neutron absorber material is Boraflex which is placed on all four sides of the storage box, and held in line with the active portion of the stored assembly by a thin sheet of steel that is welded in place. The compartments in the storage racks containing the Boraflex are exposed to the spent fuel pool environment at the top and bottom of each storage assembly. The water chemistry in the Fort Calhoun spent fuel pool has been reviewed elsewhere and found to meet NRC specifications. Type 304 stainless steel rack modules have been welded and inspected by nondestructive examinations performed in accordance with the applicable provisions of ASME Boiler and Pressure Vessel Code, Section IX. The licensee will perform a materials compatibility monitoring program consisting of 30 coupons which duplicate the conditions of Boraflex which is encased in the poison canisters. These coupons are to be mounted in an assembly which can be placed in a fuel storage position in the high density fuel racks so as to be subjected to the maximum neutron, gamma, and heat fluxes. Sufficient coupons are included to permit destructive examination of a sample on inspection intervals of 1 to 5 years over the life of the facility.

2.5.2 Chemical Compatibility

The spent fuel pool is fabricated of materials that will have good compatibility with the borated water chemistry of the spent fuel pool. The corrosion rate of Type 304 stainless steel in this water is sufficiently low to defy our ability to measure it. Since all materials in the pool are stainless steel, no galvanic corrosion effects are anticipated. No

instances of corrosion of stainless steel in spent fuel pools containing boric acid has been observed throughout the country (Ref. 8). Boraflex has been shown to be resistant to radiation doses in excess of any anticipated in the spent fuel pool (Ref. 9). The venting of the cavities containing the Boraflex to the spent fuel pool environment will ensure that no gaseous buildup will occur in these cavities that might lead to distortion of the racks. The Codes and Standards used in fabricating and inspecting these new fuel storage racks should ensure their integrity and minimize the likelihood that any stress corrosion cracking will occur during service. The materials surveillance program spelled out by the licensee will reveal any instances of deterioration of the Boraflex that might lead to the loss of neutron absorbing power during the life of the new spent fuel racks. We do not anticipate that such deterioration will occur. This monitoring program will ensure that, in the unlikely situation that the Boraflex will deteriorate in this environment, the licensee and the NRC will be aware of it in sufficient time to take corrective action.

2.5.3 Conclusion

From our evaluation as discussed above, we conclude that the corrosion that will occur in the spent fuel pool will be of little significance during the remaining life of the unit. Components of the spent fuel storage pool are constructed of alloys which are known to have a low differential galvanic potential between them, and that have performed well in spent fuel storage pools at other pressurized water reactor sites where the water chemistry is maintained to comparable standards. The proposed materials surveillance program is adequate to provide warning in the unlikely event that deterioration of the neutron absorbing properties of the Boraflex will develop during the life of the racks. Therefore, with the selection of the materials we believe that no significant corrosion should occur in the spent fuel storage racks for a period well in excess of the 40 years design life of the unit.

We, therefore, conclude that the compatability of the materials and coolant used in the spent fuel storage pool is adequate based on tests, data, and actual service experience in operating reactors. We find that the materials meet the requirements of 10 CFR Part 50, Appendix A Criterion 61, by having a capability to permit appropriate periodic inspection and testing of components, and Criterion 62, by preventing criticality by maintaining structural integrity of components and is, therefore, acceptable.

2.6 Spent Fuel Pool Cleanup System

The spent fuel pool (SFP) cleanup system is part of the pool cooling system. It consists of a demineralizer and a filter, and the required piping, valves, and instrumentation. There is also a separate skimmer system to remove surface dust and debris from the SFP. This cleanup system is similar to such systems at other nuclear plants which maintain concentrations of radioactivity in the pool water at acceptably low levels.

We expect no increase in radioactivity released to the pool water as a result of the proposed modification, and we therefore conclude the spent fuel pool cleanup system is adequate for the proposed modifications and will keep the concentrations of radioactivity in the pool water to acceptably low levels.

2.7 Occupational Radiation Exposure

We have evaluated the radiation protection aspects of the licensee's plans to modify the spent fuel pool. The licensee has estimated that 3 person-rems will be the collective occupational dose for the entire pool modification. This estimate was based on the licensee's detailed breakdown of occupational dose for each phase of the modification. In making this estimate, the licensee considered the number of individuals performing this job, their occupancy time while performing the job, and the average dose rate in the area where the job is being performed.

The pool modification consists of replacing the presently installed fuel storage racks with the new high density racks by lifting out the present ones and lowering the new ones in place. The licensee intends that the present racks will be removed, decontaminated, packaged, and shipped off site for burial as low level waste at a licensed disposal site. The licensee estimated that a total dose for decontaminating the 21 racks would be 1.89 person-rems.

A level indicator is provided within the pool to monitor spent fuel pool water level and alarms both locally and in the control room upon reaching a high level setpoint. The spent fuel pool (SFP) contains a curb or lip at the top of the SFP which will preclude minor overflowing or spilling onto the floor.

The licensee has estimated the potental increase in dose to personnel due to the SFP modification. The dose rate in the pool area due to the concentrations of radionuclides in the pool water is estimated to be 2 mr/hr. The spent fuel assemblies themselves contribute a negligible

amount of dose rates in the pool area because of the depth of water shielding in the pool. The licensee does not expect any significant increase in potential dose rate due to the buildup of crud along the sides of the pool. If crud buildup eventually becomes a major contributor to pool doses, measures will be taken to reduce such doses. The purification system for the pool includes filter and demineralizer to remove crud and will be operating during the modification of the pool.

Some of the actions that will be taken by the licensee to assure that occupational doses during each task of the pool modification will be ALARA are:

1. A health physicist will be present at all times during the re-racking to monitor for excessive airborne or high radiation by utilizing portable or hand-held radiation monitoring instruments.

2. Area radiation monitors will be used to alarm on a high radiation signal. Actual dose rates can be read locally and in the control room.

3. Personnel shall be required to wear appropriate protective clothing, as determined by the health physicist to preclude contamination.

4. As the racks are pulled out of the water, they will be washed.

5. All rack decontamination areas will be enclosed to reduce airborne contamination.

6. All personnel will be required to undergo 2 days of radiation protection training.

Based on our review of the Fort Calhoun SFP modification description and relevant experience from other operating reactors that have performed similar modifications, the staff concludes that the licensee's modification can be performed in a manner that will maintain doses to workers as low as is reasonably achievable.

We have estimated the increment in occupational dose during normal operations, after the pool modification, resulting from the proposed increase in stored fuel assemblies. The spent fuel assemblies contribute a negligible amount to area dose rates. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation dose. The small increase in radiation dose should not affect the licensee's ability to maintain individual occupational doses within the limits of 10 CFR Part 20, and as low as is reasonably achievable.

2.8 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, dated August 9, 1972. There will be no change in the waste treatment system or in the conclusions given in Section 3.1.7 and 3.1.8 of the evaluation of these systems because of the proposed modification. Our evaluation of the SFP cleanup system, in light of the proposed modification, has concluded that any resultant additional burden on the system is minimal and therefore the existing SFP cleanup system is adequate for the proposed modification and will keep the concentrations of radioactivity in the pool water within acceptably low levels.

Our evaluation of the radiological considerations supports the conclusion that the proposed modification to the Fort Calhoun, Unit No. 1, SFP is acceptable because:

(1) the conclusions of the evaluation of the waste treatment systems, as found in the Fort Calhoun, Unit 1, Safety Evaluation Report of 1972, are unchanged by the modification of the SFP; and

(2) The existing SFP cleanup system is adequate for the proposed modification.

2.9 Radiological Consequences of Cask Drop, Spent Fuel Pool Gate Drop, and Fuel Handling Accidents

2.9.1 Introduction

We have reviewed the licensee's submittal for the expansion of the storage capacity of the SFP at the Fort Calhoun Station as it relates to the above subject accidents. The review was conducted according to the guidance of Standard Review Plan 15.7.4, NUREG-0612, and NUREG-0554 with respect to accident assumptions.

2.9.2 Evaluation and Findings

CASK DROP ACCIDENTS

The licensee states that it does not presently utilize or own a spent fuel sample cask or spent fuel shipping cask. If such casks were to be utilized, the single failure-proof main hook of the Auxiliary Building Crane would be used to transport these casks. This characteristic, in conjunction

with the presence of electrical interlocks which preclude travel of the main hook and auxiliary hook over the SFP (NUREG-0612), would greatly reduce the likelihood of occurrence of a cask drop, obviating the need for consideration of radiological release impact of such an accident. Therefore, we conclude that an analysis of the radiological consequences of a cask drop accident is not required.

SPENT FUEL POOL GATE DROP ACCIDENTS

The licensee states that the non-single-failure-proof 10-ton auxiliary hook of the Auxiliary Building Crane is used to move the SFP gate. The gate is moved directly from its notch in the SFP wall to storage in the transfer canal without moving over spent fuel. If the gate were dropped and were to undergo a highly unlikely rotational motion following swinging of the supporting wire rope and hook, it could impact the tops of the spent fuel racks. The impact kinetic energy transmitted to a particular cell is no more than that transmitted in a fuel handling accident, therefore off-site radiological consequences would not exceed those from such an accident.

FUEL HANDLING ACCIDENTS

The licensee has proposed to expand the storage capacity of the SFP from 483 spent fuel assemblies to 729 assemblies. During the action, the maximum weight of loads which may be transported over spent fuel in the pool will be limited to that of a single assembly by an excess weight interlock on the fuel lifting hoist (LCO 2.8, Amendment No. 24 to Fort Calhoun Station FSAR, June 6, 1977). The proposed SFP modification does not, therefore, increase radiological consequences of fuel handling accidents considered in the staff Safety Evaluation of April 27, 1979, since the accident would still result in, at most, release of the gap activity of one fuel assembly due to the limitations on available impact kinetic energy.

2.9.2 Conclusions

Based upon the above evaluation, we conclude that the likelihood of a cask drop accident resulting in radionuclide releases is sufficiently small that this accident need not be considered. Also, if a highly unlikely SFP gate drop accident should occur, we believe that radionuclide releases no greater than those postulated in a fuel handling accident would result. Additionally, a fuel handling accident would not be expected to result in radionuclide releases leading to offsite radiological consequences exceeding those of the fuel handling accident in the staff Safety Evaluation of April 27, 1979. Our present analysis indicates a 0-2 hr. Exclusion Area Boundary (EAB) thyroid dose of 55 Rem and Whole Body dose of 0.2 Rem, given an atmospheric transport and

diffusion Relative Concentration value of 2.8×10^{-4} sec/m^3. These conservatively estimated doses are well within the 10 CFR Part 100 guideline values. Therefore, we conclude that the proposed modifications are acceptable.

3.0 Conclusions

Based on our review, we conclude that the licensee's proposed SFP modification which will entail installing new racks with a total capacity of 729 whole fuel assemblies is acceptable. In addition, the proposed TS changes for Fort Calhoun are acceptable.

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 the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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