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

SAFETY EVALUATION REPORT

RELATED TO AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NPF-9

AND TO AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

INTRODUCTION

By letter dated February 17, 1984, Duke Power Company (the licensee) made application to amend the operating licenses for the McGuire Nuclear Station, Units 1 and 2, to allow an expansion of the spent fuel pool storage capacity at each unit from 500 to 1463 storage spaces. The proposed expansion is to be achieved by reracking each spent fuel pool with two-region poisoned racks. This expansion will provide storage until 2010 with space for offloading a full core assuming reloads of a third of a core. The licensee identified in the FSAR and in the spent fuel pool storage capacity submittal that spent fuel from the Oconee reactor may be stored in the McGuire spent fuel pool. The storage of Oconee fuel at McGuire was evaluated and reported in Supplement No. 2 of the Safety Evaluation Report (NUREG-0422). Our evaluation of the licensee's capacity expansion submittal does not change the conclusion in the SER Supplement No. 2 related to the storing of Oconee spent fuel at McGuire.

EVALUATION

In order to support this amendment application, the licensee by letter dated March 20, 1984, submitted a "Spent Fuel Pools Rerack Modification Safety and Environmental Analysis" which served as a keystone for the staff evaluation. Supplemental information was provided by the licensee as reflected in the following evaluation summary.

1.0 Criticality Aspects

Description of Racks

Each pool will contain racks that provide 1463 designated locations for the storage of reactor fuel. The storage racks will be divided between two regions - one containing 286 locations and one containing 1177. The smaller region, having sufficient capacity for approximately 1½ full cores, will be used for the storage of fresh fuel and fuel not suitable for Region 2. The larger region will normally be restricted to fuel having a specified minimum burnup. The licensee proposes that, if at some future date Region 1 becomes filled, storage of high reactivity fuel (up to fresh 4.0 percent enrichment) be permitted in Region 2 in a checkerboard array with every other location empty. Physical barriers will be used to prevent storage in the empty locations.

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The Region 1 racks will consist of stainless steel cans of 8.75 inches square interior dimension and 0.075 inch wall thickness. On the outer surface of each side of the cans Boraflex sheets having a minimum areal density of 0.02 gram per square centimeter of B-10 are held in place by a thin-walled stainless steel wrapper plate. The rack structure maintains these cans on a 10.40 inches center-to-center spacing.

The Region 2 rack design consists of stainless steel cans welded together to form a honeycomb type structure. The cans have an interior square dimension of 8.93 inches and are made of stainless steel. All four sides of interior cans have Boraflex sheets containing 0.006 gram of B-10 per square centimeter of surface area that are held in place by a stainless steel wrapper which is spot welded to the can. The resulting structure maintains the stored fuel assemblies at a center-to-center spacing of 9.125 inches.

On the outer boundary of each rack the Boraflex sheet is omitted. Neighboring racks are maintained at sufficient separation from each other to preclude an increase in pool reactivity from this cause.

Calculation Methods

The calculation of the effective multiplication factor, k-eff, for Region 1 makes use of the AMPX system of codes for cross-section preparation and the Monte-Carlo Code KENO-IV for reactivity. This code set has been verified against a set of 27 critical experiments that simulate various features of the rack design. A calculational method bias of zero and uncertainty of 0.013 k-eff (95/95) was inferred from these comparisons.

The calculation of the criterion for acceptable burnup for storage in Region 2 makes use of the concept of reactivity equivalence. Since the KENO-IV code cannot handle burned fuel assemblies, it is necessary to obtain the fresh fuel assembly enrichment which yields the same pool k-eff as the burned assembly. Because of the presence of the poison in the Region 2 racks a multigroup transport theory code is more appropriate than diffusion theory for this calculation. The PHOENIX code was used.

The calculation proceeds as follows:

1. An end-point of 36.5 GWD/MT burnup for a bundle having an initial enrichment of 4.0 weight percent U-235 is chosen

2. PHOENIX is used to calculate the k_∞ of such an assembly in the rack geometry (including can and Boraflex absorber)

3. The burnup required to produce the same k_∞ is calculated for a number of smaller enrichments

4. The enrichment required to produce the same k without burnup is obtained (in the present case the value is 1.4 percent)

5. KENO-IV is used to calculate the rack multiplication factor for the 1.4 percent enrichment assembly.

The advantage of this procedure is that only relative multiplication factors are computed by PHOENIX. The final value of the rack multiplication factor is obtained from the more powerful KENO-IV code.

Treatment of Uncertainties

For the Region 1 analysis the total uncertainty is the statistical combination of the method uncertainty, the uncertainty in the particular KENO calculation, and mechanical uncertainties due to tolerances, spacing, etc. The mechanical uncertainties were treated either by making worst case assumptions (e.g., using the minimum rather than nominal value of the boron loading) or by performing sensitivity studies and obtaining a value for the uncertainty in rack multiplication factor due to uncertainty in dimensions.

In the Region 2 analysis the same uncertainties are considered along with others that are unique to the rack design and usage. These include uncertainty due to particle self-shielding in the boron (actually bias), uncertainty in the plutonium reactivity and uncertainty in the reactivity as a function of burnup. Including both the plutonium and burnup reactivity uncertainties is conservative since the latter includes the former as one of its components. The particle self-shielding bias is important for Region 2 because of the low boron loading relative to Region 1 (0.006 vs. 0.02 gm/cm^2 of B-10).

The PHOENIX code was qualified for burnup calculations by comparing calculated isotopic ratios to measurements made in Yankee-Rowe Core 5, and by comparison of equivalent reactivity burnup between PHOENIX and the LEOPARD/TURTLE code. A set of 81 critical experiments was analyzed to qualify the code for zero burnup conditions. Conservative uncertainties of 5 percent of the reactivity worth of the actinides and 5 percent of the reactivity change due to burnup have been assigned to these parameters.

Results of Analysis

Normal Storage

For Region 1, the rack multiplication factor is calculated to be 0.944, including uncertainties at the 95/95 level, when fuel having an enrichment of 4.0 weight percent U-235 is stored therein. Fuel of either the

Westinghouse standard or OFA design may be stored. Pure water at 1.0 gram per cubic centimeters is assumed.

For Region 2, the rack multiplication factor is 0.940 for the most reactive irradiated fuel permitted to be stored in the racks, i.e., fuel with the minimum burnup permitted for each initial enrichment. For fresh fuel (4.0 percent enrichment) stored in a checkerboard array in the racks, the effective multiplication factor is 0.866. These multiplication factors include all uncertainties and are obtained for pure water at a density of one gram per cubic centimeter. Burned fuel of the Westinghouse standard or OFA design or of the Babcock and Wilcox 15x15 design may be stored in Region 2. Analyses were performed for all three fuel types and the proposed curve of burnup vs. initial enrichment bounds the results of the calculation.

Abnormal Storage Conditions

Most abnormal storage conditions will not result in an increase in k-eff of the racks. For example, loss of a cooling system will result in an increase in pool temperature but this causes a decrease in the k-eff value. It is possible to postulate events (e.g., a seismic event) which could lead to an increase in pool reactivity. However, for such events, credit may be taken for the approximately 2000 ppm of boron in the pool water. The reduction in the k-eff value caused by the boron (approximately 0.25) more than offsets the reactivity addition caused by credible accidents.

Summary

The following discussion summarizes our evaluation of the proposed re-racking of the McGuire spent fuel storage pools related to criticality aspects.

We have reviewed the assumptions made in the performance of the criticality analyses. These include use of the highest permitted reactivity bundle, pure water moderator at a density of 1.0 gram per cubic centimeter, and an infinite array of assemblies. These are consistent with NRC guidelines and are acceptable.

We have reviewed the uncertainties which have been included. For Region 1 these include variation in poison pocket thickness, stainless steel thickness, cell interior dimensions, center-to-center spacing and cell bowing. Other parameters, such as boron loading, are taken at their most conservative limits. For Region 2 additional uncertainties due to burnup calculations and calculations of plutonium worth are included. For both regions calculational uncertainties and biases are included. These uncertainties meet our requirements and are acceptable.

We have reviewed the verification of the calculational methods. The KENO-IV code is widely used in the industry for the purpose of calculating fuel rack criticality. The set of benchmark critical experiments used to verify the calculational method encompasses the enrichment, separation distance and separating material used in the racks. The set of experiments used to verify the PHOENIX code for the reactivity equivalence calculations is adequate and encompasses the pellet size and enrichment of the fuel proposed for storage in the McGuire racks. The uncertainties in the burnup and plutonium worth are verified against Yankee Core 5 isotopics and comparisons with the Westinghouse design LEOPARD/TURTLE code package. We find that adequate verification of the codes used in the criticality analyses has been performed.

The technique of using reactivity equivalencing to define the storage criterion (burnup as a function of initial enrichment) is, in some form, in widespread use in the industry and is acceptable.

For Region 1 racks we have compared the results of the McGuire calculation to a generic study and found them to be compatible. Finally the results of the calculation for Region 1 and 2 meet our acceptance criterion of less than or equal to 0.95 including all uncertainties at the 95/95 level.

We have reviewed the proposed Technical Specification 3/4.9.12 and find that it is consistent with the assumptions in the safety analysis and is acceptable.

Conclusions

Based on our review, which is described above, we find the criticality aspects of the design of the spent fuel racks to be acceptable. We conclude that fresh Westinghouse 17x17 fuel of either the standard or OFA design may be safely stored in Region 1 so long as its enrichment does not exceed 4.0 w/o U-235. We further conclude that either type of Westinghouse fuel or fuel of the 15x15 Babcock and Wilcox design may be stored in Region 2 provided it falls in the acceptable region of Figure 2.4-3 of the Safety and Environmental Analysis report. Fuel which does not meet this criterion may be stored in Region 2 provided it is stored in a checkerboard arrangement with every other location vacant. The licensee has committed to provide physical barriers to prevent storage in the empty locations.

2.0 Systems Aspects

Decay Heat Loads

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 18.0 MBTU/Hr. This heat load results in a maximum bulk pool temperature of 133°F. The expected maximum abnormal heat load following a full core discharge is 41.6 MBTU/Hr. This abnormal heat load results in a maximum bulk pool temperature of 178°F, or a maximum bulk pool temperature of 120°F with both cooling trains operating. Assuming the loss of all cooling, boiling would occur after 13.8 hours for the normal heat load condition and after 4.7 hours for the maximum heat load condition. This provides reasonable time to initiate makeup to the spent fuel pool.

The spent fuel pool water is cooled by the component cooling water system, which in turn is cooled by the service water system. The licensee proposed no modifications to these two systems as part of this spent fuel pool expansion project. Although the spent fuel pool heat exchanger only has a design capacity of 15.0 MBTU/Hr, an independent pool water temperature calculation was performed which verified that the pool water temperature will remain within acceptable limits. Thus our review of these systems as to their adequacy to remove the additional heat load indicates that they are capable of removing the additional heat.

Control of Heavy Loads

Presently, there is no spent fuel in the McGuire Unit 2 spent fuel pool. There is spent fuel in the McGuire Unit 1 spent fuel pool and the licensee has stated that no spent fuel racks will be carried over spent fuel. However, the empty spent fuel storage racks will be carried over an empty portion of two racks which will contain spent fuel. The utility has performed a load drop analysis which indicated that there would be no damage to the stored fuel and therefore no radiological consequences. The postulated rack drop would not change the separation distance between the stored fuel assemblies or the concentration of boron. Therefore, the margin of safety to criticality will not be affected by a rack drop accident. The racks will enter and exit the fuel building by means of the outdoor cask handling crane. The racks will be maneuvered inside the fuel building by means of the fuel building crane. There are no safety related components on or above the fuel handling floor. Therefore, a drop of a spent fuel rack will not have any adverse consequences as identified in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and is, therefore, acceptable.

Summary and Conclusions

Based on the above, we conclude that the proposed overall spent fuel pool storage capacity modification program is acceptable for storage of 1463 spent fuel assemblies per reactor unit with respect to the storage rack capacity, the developed heat loads and pool water temperatures, the load handling, and the spent fuel pool cooling and support system capabilities.

3.0 Material Considerations

Description

The safety function of the spent fuel pool and storage rack system is to maintain the spent fuel assemblies in a subcritical array during all credible storage conditions. We have reviewed the compatibility and chemical stability of the materials, except the fuel assemblies, wetted by the pool water.

The spent fuel racks in the proposed expansion would be constructed entirely of Type 304 stainless steel, except for the nuclear poison material. The existing spent fuel pool liner is constructed of stainless steel. The high density spent fuel storage racks will utilize Boraflex[1] sheets as a neutron absorber. Boraflex consists of boron carbide powder in a rubber-like silicone polymeric matrix. The spent fuel storage rack configuration is composed of individual storage cells interconnected to form an integral structure. The major components of the assembly are the fuel assembly cells, the Boraflex material, the wrapper and the upper and lower spacer plates.

The upper end of the cell has a funnel shape flare for easy insertion of the fuel assembly. The wrapper surrounds the Boraflex material, but is open at the top and bottom to provide for venting of any gases that are generated. The Boraflex sheets sit in a square annular cavity formed by the square inner stainless steel tube and the outer wrapper. Each sheet is supported by lower spacer plate.

Corrosion & Materials Compatibility

The pool liner, rack lattice structure and fuel storage tubes are stainless steel which is compatible with the storage pool environment. In this environment of oxygen-saturated borated water, the corrosive deterioration of the Type 304 stainless steel should not exceed a depth of 6.00×10^{-5} inch in 100 years, which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar potentials. The Boraflex is composed of non-metallic materials and therefore will not develop a galvanic potential in contact with the metal components. Boraflex has undergone extensive testing to study the effects of gamma irradiation in various environments, and to verify its structural integrity and suitability as a neutron absorbing material. The evaluation tests have shown that the Boraflex is unaffected by the pool water environment and will not be degraded by corrosion. Tests were performed at the University of Michigan,[2] exposing Boraflex to 1.103×10^{11} rads of gamma radiation with substantial concurrent neutron flux in borated water. These tests indicate that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of borated water and gamma irradiation. Irradiation will cause some loss of flexibility, but will not lead to break up of the Boraflex. Long term borated water soak tests at high temperatures were also conducted.(3) The tests show that Boraflex withstands a borated water immersion of 240°F for 260 days without visible distortion or softening. The Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide.

The annulus space which contains the Boraflex is vented to the pool at each storage tube assembly. Venting of the annulus will allow gas generated by the chemical degradation of the silicone polymer binder during heating and irradiation to escape, and will prevent bulging or swelling of the inner stainless steel tube.

The tests(1) have shown that neither irradiation, environment nor Boraflex composition has a discernible effect on the neutron transmission of the Boraflex material. The tests also show that Boraflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions are reached regarding the leaching of elemental boron from the Boraflex. Boron carbide of the grade normally in the Boraflex will typically contain 0.1 wt percent of soluble boron. The test results have confirmed the encapsulation function of the silicone polymer matrix in preventing the leaching of soluble specie from the boron carbide.

To provide added assurance that no unexpected corrosion or degradation of the materials will compromise the integrity of the racks, the licensee has committed to conduct a long term fuel storage cell surveillance program. Surveillance samples are in the form of removable stainless steel clad Boraflex sheets, which are proto-typical of the fuel storage cell walls. These specimens will be removed and examined periodically (approximately 5 year intervals).

Summary and Conclusion

From our evaluation as discussed above we conclude that the corrosion that will occur in the spent fuel storage pool environment should be of little significance during the life of the plant. Components in the spent fuel storage pool are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion. Tests under irradiation and at elevated temperatures in borated water indicate that the Boraflex material will not undergo significant degradation during the expected service life.

We further conclude that the environmental compatibility and stability of the materials used in the expanded spent fuel storage pool is adequate based on the test data cited above and actual service experience in operating reactors.

We have reviewed the surveillance program and we conclude that the monitoring of the materials in the spent fuel storage pool, as proposed by the licensee, will provide reasonable assurance that the Boraflex material will continue to perform its function for the design life of the pool. 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.

We, therefore, find that the implementation of a monitoring program and the selection of appropriate materials of construction by the licensee meet the requirements of 10 CFR Part 50, Appendix A, Criterion 61, having a capability to permit appropriate periodic inspection and testing of components, and Criterion 62, preventing criticality by maintaining structural integrity of components and of the boron poison and is therefore acceptable.

References

1. J. S. Anderson, "Boraflex Neutron Shielding Material -- Product Performance Date," Brand Industries, Inc., Report 748-30-1, (August 1979).

2. J. S. Anderson, "Irradiation Study of Boraflex Neutron Shielding Materials," Brand Industries, Inc., Report 748-10-1, (August 1981).

3. J. S. Anderson, "A Final Report on the Effects of High Temperature Borated Water Exposure on BISCO Boraflex Neutron Absorbing Materials," Brand Industries, Inc., Report 748-21-1, (August 1978).

4. Structural Considerations

General

The structural review of the proposed spent fuel storage pool expansion was performed by our consultant, the Franklin Research Center (FRC). The results of their review are described in the FRC Technical Evaluation Report TER-C5506-526, revised August 10, 1984, which is incorporated by reference in this safety evaluation.

The spent fuel pools are constructed of reinforced concrete lined with stainless steel plates. The Unit 1 pool is a flat, reinforced concrete slab 4.5 feet thick, supported on reinforced concrete beams, 6.5 feet deep by 4.5 feet wide, and by 4.0 feet thick walls at the perimeter. The beams are spaced on 20.0 foot centers, spanning across the pool in the short direction. Floor loads are transmitted to the structural foundation from the floor slab, to the deep beams, to the perimeter walls, then to the bedrock foundation. The Unit 2 pool floor is supported continuously on a bedrock foundation. The floor slab is 11.0 feet thick. The additional thickness as compared to Unit 1 is due to construction considerations. All dead, live and seismic loads are transmitted directly through the floor slab then to the bedrock foundation.

Fuel storage is divided into two regions within each pool. The Region 1 storage racks are composed of individual storage cells made of stainless steel. The cells within a module are interconnected by grid assemblies to form an integral structure. Each rack module is provided with leveling pads which contact the spent fuel pool floor and are remotely adjustable from above throughout the cells at installation. The modules are free-standing and are not anchored to the floor nor braced to the pool walls. The fuel rack assembly consists of three major sections which are the leveling pad assembly, the lower and upper grid assemblies, and the cell assembly.

The Region 2 storage racks consist of stainless steel cells assembled in a checkerboard pattern, producing a honeycomb type structure. The cells are welded to a base support assembly and to one another to form an integral structure without use of grids as used in Region 1 racks. This design is also provided with leveling pads which contact the spent fuel pool floor and are remotely adjustable from above through the cells at installation. The modules are free standing and are not anchored to the floor nor braced to the pool walls. The fuel rack module consists of two major sections which are the base support assembly and the cell assembly.

<u>Codes, Standards and Design Loads</u>

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. The existing concrete pool structure was evaluated for the new loads in accordance with the requirements of the McGuire FSAR Section 3.8.4. 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.

Seismic loads for the rack design are based on the original design floor acceleration response spectra calculated for the plant at the licensing stage. The seismic loads were applied to the model in three orthogonal directions. Loads due to a fuel bundle drop accident were considered in a separate analysis. The postulated loads from these events were found to be acceptable.

Design and Analysis Procedures by Licensee

a. Design and Analysis of the Racks

The dynamic response and internal stresses and loads are obtained from a seismic analysis which is performed in two phases. The first phase is a time history analysis on a simplified nonlinear finite element model. The second phase is a response spectrum analysis of a detailed linear three dimensional rack assembly finite element model. Two percent damping is used in the seismic analysis for both the OBE and SSE. Further details on the methodology is discussed in Franklin Research Technical Evaluation Report TER-C5506-526, revised August 10, 1984.

Calculated stresses for the rack components were found to be 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 also 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 load in the fuel assembly. Resulting stresses were found to be acceptable.

b. Analysis of the Pool Structure

The slab, beams and walls are reinforced to meet all FSAR criteria. The existing structures were analyzed for the modified fuel rack loads using the STRUDL finite element computer program. Original plant response spectra and damping values were used in consideration of the seismic loadings. Design criteria, including loading combinations and allowable stresses, are in compliance with McGuire FSAR Section 3.8.4 and the existing spent fuel pools are determined to safely support the loads generated by the new fuel racks.

Conclusion

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

5.0 Radiation Protection

Occupational Radiation Exposure

The occupational exposure estimated by the licensee for the rerack modification is 15 person-Rem. This estimate is based on a detailed breakdown of occupational exposure for each phase of the rerack modification, and considers the number of individuals performing a specific job, the average dose rates in the area where the job is being performed, and the time spent by workers performing the job in these areas. This estimate represents a small fraction of the annual dose estimated for the station - less than 1.5%. The rerack modification for Unit 2 will involve no occupational exposure, since the operation will be carried out in a dry, radiologically "clean" pool.

During the modification process and during projected operations, the fuel assemblies themselves (including the additional assemblies) will contribute a negligible amount to pool area dose rates due to the depth of water shielding the fuel.

Radioactive activation and corrosion products (crud) may be released to the pool water from fuel surfaces during fuel movements during the modification. This could increase radiation levels in the vicinity of the pool, however, the Spent Fuel Pool Cooling System, in conjunction with water vacuuming of the pool floor, walls and fuel rack surfaces, will filter and purify the pool water. This will remove the crud and minimize the dose contribution from crud in the pool water to workers and divers in the pool area.

The licensee has considered burial, decontamination, and long-term on-site storage as means to dispose of old racks from the Unit 1 Spent Fuel Pool. Following removal from the pool, the racks will be rinsed by low pressure spray or hydrolased to reduce contamination levels and subsequent handling doses. Protective clothing and respiratory protection will be utilized as needed to keep exposures to contamination and airborne radioactivity ALARA. The racks will be decontaminated, if possible, and sold as scrap or shipped to a burial site if decon is not practical. Unit 2 racks will be radiologically clean and will be sold as scrap.

Doses to divers will be minimized by rearrangement of stored assemblies to give the lowest practical dose rates. Additionally, diver paths will be marked, and health physics personnel will monitor their work. Radiation protection controls for divers include protective clothing, multiple - TLD's and self-reading dosimeters, underwater surveys, management dose tracking, and direct communications with divers. Divers will be warned if they approach high radiation/exclusion zones, and will be kept at least 10 feet from spent fuel.

Periodic radiation and contamination surveys will be conducted in work areas, and grab sampling and/or continuous sampling performed where there is a potential for airborne radioactivity.

Work will be controlled by Radiation Work Permit, posting, use of stay time, zoning, access control, and health physics personnel to assure that doses are kept ALARA.

Additionally, we have estimated the increment in onsite occupational dose during normal operations after the pool modification as a result of the proposed increase in stored fuel assemblies. This estimate is based on information supplied by the licensee 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. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent of the total annual occupational radiation exposure at both units. The small increase in radiation exposure should not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable levels and within the limits of 10 CFR Part 20.

Conclusion

Based on the manner in which the licensee will perform the modification; our previous evaluation of their radiation protection/ALARA program during the licensing process; the radiation protection measures proposed for the modification task, including radiation, contamination, and airborne radioactivity monitoring; and relevant experience from other operating reactors that have performed similar spent fuel pool modifications, the staff concludes that adequate radiation protection measures have been taken to assure worker protection, and the McGuire Spent fuel pool modification can be performed in a manner that will ensure that doses to workers and the general public will be ALARA and that storing additional fuel in the two pools will not result in any significant increase in doses received by workers.

SAFETY CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (49 FR 27225) on July 2, 1984, and consulted with the state of North Carolina. No public comments were received, and the state of North Carolina did not have any comments.

In conclusion the staff finds the proposed changes to the plant technical specifications to be acceptable and 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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AMENDMENT NO. 35 TO FACILITY OPERATING LICENSE NPF-9 - McGUIRE NUCLEAR STATION, UNIT 1
AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE NPF-17 - McGUIRE NUCLEAR STATION, UNIT 2

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