UNITED STATES
NUCLEAR REGULATORY COMMISSION
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DESIGNATED ORIGINAL

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SAFETY EVALUATION BY THE OFFICE OF
NUCLEAR REACTOR REGULATION

RELATING TO THE MODIFICATION OF THE
SPENT FUEL STORAGE POOL

FACILITY OPERATING LICENSE NO. DPR-23
CAROLINA POWER AND LIGHT COMPANY
H.B. ROBINSON STEAM ELECTRIC PLANT UNIT 2

DOCKET NO. 50-261

Contents

1.0 INTRODUCTION

By letter dated December 1, 1980, as supplemented April 10, May 11, June 15, June 18, August 28, 1981, and April 2, 1982, Carolina Power and Light Company (CP&L or the licensee) requested an Amendment to Facility Operating License No. DPR-23 for H.B. Robinson Steam Electric Plant Unit 2 (Robinson Unit 2). The request would revise the Radiological Technical Specifications to allow an increase in the spent fuel pool (SFP) storage capacity from 276 to a maximum of 544 fuel assemblies through the use of neutron absorbing "poison" spent fuel storage racks.

The expanded storage would allow Robinson Unit 2 to operate until 1986 with capability for a full core discharge, assuming annual one-third core reloads.

The major safety considerations associated with the proposed expansion of the Robinson Unit 2 SFP storage capacity are addressed below. A separate Environmental Impact Appraisal has been prepared as part of this licensing action.

2.0 BACKGROUND

The H.B. Robinson Unit 2 spent fuel pool currently contains racks with a capacity of 276 fuel assemblies. These racks include 36 cells that were installed in 1976. The proposed modification will add 368 high density cells and remove 100 existing cells including those installed in 1976. A net increase of 268 storage spaces will bring the total storage capacity to 544 spaces; however, 10 spaces will be administratively controlled as unused spares.

The new cell assemblies are made of stainless steel and enclosed with sheets of a neutron absorbing material, Boraflex (0.02 gram of Boron-10 per square centimeter). A stainless steel wrapper is spot welded to the cell to provide a cover for the Boraflex material. The cells have inside dimensions of 8.75 inches and a center-to-center spacing of 10.5 inches. Three of the new modules will have a 12x8 fuel assembly configuration and the other new module will be a 10x8 configuration. Eleven of the current 4x4 rack modules will remain in the spent fuel pool. The racks are to be free-standing within the pool with a minimum separation between new racks of one inch and a separation between new and existing racks of about six inches.

Removal of a portion of the existing fuel storage racks (four 4x4 and four 3x3 modules) and installations of the new High Density Fuel Storage System modules will be accomplished by shuffling the spent fuel without emptying the pool. A temporary hoist will be provided to move the old and new racks in a sequence that will prevent transporting loads over stored spent fuel. Divers will be employed to work where underwater access is required. It is estimated that 75 working days will be required to complete the work at an estimated cost of $3,427,000.

3.0 DISCUSSION AND EVALUATION

3.1 Criticality Considerations

The licensee has provided an analysis of the criticality of the proposed storage racks. This analysis was made under the assumption of an enrichment of 3.9 w/o U-235, the maximum value authorized for storage. Pure water at a density of 1.0 gm/cm^3 was assumed and the array was assumed to be infinite in lateral and axial extent. Credit was taken for neutron absorption in only the stainless steel can and wrapper and for the Boraflex absorber. Our evaluation has assumed the presence of this absorber.

The calculations were performed with the KENO-IV Monte-Carlo code with cross sections prepared by the AMPX system of codes including the NITAWL and XSDRNPM codes. This combination is widely used and is acceptable. Use by Carolina Power and Light has been verified by comparison of code results with a series of 27 experiments. The experiments encompass the Robinson enrichment value and included stainless steel and boron absorber plates. The comparison showed a zero value of bias and an uncertainty of 0.013 reactivity change at the 95 percent probability and 95 percent confidence level. This value was combined with the statistical uncertainty in the Monte-Carlo calculation to obtain the total calculational uncertainty.

Mechanical uncertainties were treated for the most part by using limiting conservative values for the input values to the codes. A specific bias was included to account for particle self-shielding in the Boraflex sheet.

The effective multiplication factor for the racks, including all uncertainties, is 0.924 which meets our acceptance criterion of 0.95.

The effect of credible accidents has been considered. Loss of cooling will result in a decrease in reactivity. A fuel assembly lying on top of the racks is sufficiently separated from the fuel in the racks to cause a negligible increase in reactivity. Other accidents considered included damage to the racks from misuse of the lifting crane, and damage due to dropping an assembly onto the racks. Even without credit for the soluble boron in the pool none of these events violates the 0.95 acceptance criterion. If the double contingency principle is evoked such credit may be taken and the multiplication factor is reduced from the value for the nominal racks.

We find the criticality aspects of the proposed high density fuel racks to be acceptable. This finding is based on the following considerations:

1. the input parameters assumed for the design are conservative;

2. the calculations are done by a widely used state-of-the-art method;

3. the calculation method has been verified by comparison with experiment;

4. uncertainties in the calculations and input conditions have been considered; and

5. our acceptance criterion is met with ample margin.

3.2 Spent Fuel Cooling

The licensee's December 1, 1980 submittal presented the calculated maximum decay heat load, assuming normal discharges and normal discharges plus a full core discharge, for the pool's present storage capacity as well as the future heat loads following the expansion. The calculations were performed in accordance with ANS 5.1, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors" and NRC Branch Technical Position APCSB 9-2. The H.B. Robinson core consists of 157 fuel assemblies, and in calculating the decay heat load, it was assumed that the normal annual discharges (i.e., 1/3 of a core) had been irradiated at the licensed thermal power level of 2300 mwt. The results are presented in Table 1.

Table 1 Decay heat generated by spent fuel

	Total number fuel assembly	Decay time hours	Calculated heat load x 106 btu/hr	Calculated pool water temperature °F
Existing Storage Capacity				
1. Normal discharges (52 fuel assemblies)	261	118	9.5	125
2. Normal discharges plus full core discharge	261	154	24.5	162
Expanded Storage Capacity				
1. Normal discharges (52 fuel assemblies)	534	118	12.0	132
2. Normal discharges plus full core discharge	534	154	26.0	166

The existing spent fuel pool cooling system consists of a single loop having a stainless steel centrifugal pump, a shell and U-tube heat exchanger and associated stainless steel piping and valves. The loop design is such that in the unlikely event of any pipe failure the spent fuel pool water cannot be gravity drained below a level of six feet above the top of the stored spent fuel. The pool water level is continuously monitored by level instrumentation that actuate alarms in the Control Room. The water level variation between the high and low level alarms is nine inches. To prevent the loss of cooling in the event of a pump failure, a second pump will be installed in parallel with the existing pump. These two identical pumps are each rated at 2,300 gpm. They have been designed in accordance with the following requirements: ASA B16.5, NEMA Std. MG1-1963, ASTM and ASME Code Sections III, VIII and IX. Whereas it is not anticipated that a situation would arise where another pump would be needed, a third pump will be available, although not permanently installed in the pool cooling system. This pump is of normal industrial design and would be connected to the cooling system by flanged connections. The rejected heat is transferred from the pool water to the service water system via the component cooling water system.

As illustrated in Table 1, the heat removal capability of the spent fuel pool cooling system increases as the pool water temperature increases. Further, these data indicate, for the assumed decay times, that the calculated pool temperature would exceed 150°F, as stipulated in the FSAR when fuel core discharges occur. CP&L has indicated that in order to prevent the pool water temperature from exceeding 150°F during full core discharges they will administratively increase the interval over which a full core discharge takes place. In the case of the maximum calculated decay heat load described in Table 1 (i.e., expanded storage capacity and a full core discharge), CP&L indicates that the minimum time interval between shutdown and completion of full core discharge will be 13.2 days.

We have checked the decay head load following 13.2 days of decay time and find that it closely approximates the heat removal capability when the pool water temperature is 150°F. Therefore we conclude it is acceptable.

In order to provide assurance that the pool water temperature does not exceed 150°F during all full core discharges the licensee will monitor the pool water temperature after the insertion of each additional fuel assembly after the first 52 fuel assemblies have been discharged. Should the pool temperature exceed 150°F, fuel assemblies are to be transferred back to the reactor cavity until the temperature falls to 150°F or less.

The maximum calculated pool water temperature of 132°F, for normal discharges, is less than the 140°F required by Standard Review Plan 9.1.3 and is therefore acceptable.

Contingent upon the pool water temperature being monitored, as described, during full core discharges, we conclude that the spent fuel pool cooling system is adequate and therefore acceptable.

In the unlikely event that all pool cooling is lost when the pool contains its maximum heat load and the water temperature is 150°F, CP&L indicates it would take 6.83 hours for the pool water to reach a boiling temperature. Under such conditions the boil off rate, i.e., the makeup requirement, would be 41.23 gpm. The normal fuel pool makeup water source is the refueling water storage tank via the fuel pool purification pump. It is capable of 100 gpm. We have reviewed the calculations and conclude that the time interval of 6.83 hours, before boiling, is sufficient to perform the more likely types of corrective maintenance or other corrective measures and is therefore acceptable. Further, the boil off rate is consistent with the decay heat load and less than half of the possible makeup rate and is therefore acceptable.

We conclude that the single loop cooling system is adequate to handle the heat load of 534 spent fuel assemblies and that, in the event of pump failure, sufficient pump redundancy or makeup requirement is available to prevent excessive loss of water from the spent fuel pool.

3.3 Handling of Storage Racks

The H. B. Robinson spent fuel pool, a seismic Category I reinforced concrete structure containing 35,167 cubic feet of water, is housed within the Fuel Handling Building. A 125-ton capacity outside overhead crane is provided to handle heavy loads such as the spent fuel shipping cask. The range of travel of this crane is such that it can only pass over a portion of the spent fuel storage pool. Therefore the removal and installation of storage racks will require that a temporary traveling bridge and hoist be installed on the fuel handling bridge rails for the lateral movement of the storage racks to and from the spent fuel cask loading area. Since the 125-ton capacity overhead crane is an outside crane, certain Fuel Handling Building roof and wall panels must be moved in order to allow the passage of loads in and out of the building. Based on previous submittals NRC concluded in 1977 that the overhead cask handling crane met the intent of APCSB Branch Technical Position 9-1 (i.e., single-failure-proof crane) and was therefore acceptable.

The June 15, 1981 submittal indicates that the movement of all loads into and out of the Fuel Handling Building, associated with this modification, will be accomplished utilizing the single-failure-proof cask crane and/or double rigging to assure that a single failure will not result in an unanalyzed load-drop event. The rig used in handling the racks is being designed by Westinghouse. It will be a four point single-failure-proof lifting device. Redundant slings and shackles will transfer this load to the hook. During lateral movements of the storage racks within the pool, using the temporary bridge and hoist, the racks will not be lifted more than six inches above the pool floor in order to minimize the consequences of a drop. Since some of the existing storage racks have been welded to the pool bottom, divers will be required to cut the welds. In order to minimize diver radiation exposure the fuel will be moved away from the respective work areas. Therefore spent fuel will not be in or near the work area or the storage rack travel paths. Further, no loads will pass over stored spent fuel.

The August 12, 1981 submittal relating to the Control of Heavy Loads states that all crane operators and signalmen are trained in accordance with ANSI B30.2-1976, and no exceptions are taken regarding training, qualification or operator conduct.

We have reviewed the described load handling operations and the following Technical Specification restrictions and requirements:

1. Spent Fuel Cask Handling Crane Load handling operations are only permitted when the ambient outside air temperature is greater than 33°F.

2. The hoist, bridge and trolley travel limit switches of the Spent Fuel Cask Handling Crane shall be tested prior to each period of service and on a monthly basis when the crane is in service.

3. Crane ropes shall be inspected in accordance with ANSI B30.2 prior to each period of service and on a monthly basis when the crane is in service.

On the basis of our review we conclude that the licensee will use acceptable procedures and load handling equipment.

3.4 Fuel Handling

The NRC staff has underway a generic review of load handling operations in the vicinity of spent fuel pools to determine the likelihood of a heavy load impacting fuel in the pool and, if necessary, the radiological consequences of such an event. Because Robinson-2 will be required (by Technical Specification) to prohibit loads greater than the nominal weight of a fuel assembly and handling pool to be transported over spent fuel in the SFP, we have concluded that the likelihood of a load handling accident is sufficiently small that the proposed modification is acceptable, and no additional restrictions on load handling operations in the vicinity of the SFP are necessary while our generic review is underway.

The potential consequences of fuel handling accidents (i.e., rupture of fuel pins in one fuel assembly and subsequent release of the radioactive inventory within the gap) in the spent fuel pool area presented in the SE dated May 18, 1970 are not changed by the use of high density racks, since the amount of fuel damage in this accident remains unchanged.

3.5 Structural and Seismic Loadings

3.5.1 General

The Robinson spent fuel pool is located in a separate building adjacent to the containment building. The walls of the pool are six feet thick and the floor is 4.5 feet thick. The pool is lined with a water-tight stainless steel liner. A new steel column is to be installed in the space below the pool floor and above the base slab to provide additional support for the high density racks.

3.5.2 Applicable Codes, Standards, and Specifications

Structural material of the racks conforms to the ASME B&PV Code, Section III, Subsection NF. Computed stresses were compared with the ASME B&PV Code, Section III, Subsection NF. Load combinations and acceptance criteria for the racks are in accordance with 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 pool structure was evaluated in accordance with the requirements of ACI-318-63, which was the original construction document.

3.5.3 Seismic and Impact Loads

The seismic time history excitation used in the nonlinear analysis of the fuel rack assembly was developed from the ground response spectrum and damping values contained in the USNRC Regulatory Guides 1.60 and 1.61 respectively. A ground acceleration of 0.2 g's horizontal and 0.134 g's vertical was used as the basis for the SSE event. Using this information the design time history was synthesized. The time histories were verified by constructing response spectra which were shown to envelope, from above, the Regulatory Guide 1.60 response spectra. The effect of potential fuel bundle/rack impact was considered in the seismic analysis. As noted above this required consideration of nonlinear effects. Loads from a fuel drop accident were postulated. Loads from a possible crane uplift event were also considered.

3.5.4 Loads and Load Combinations

Loads and load combinations for the racks and the pool structure were reviewed and found to be in conformance with the applicable portions of "NRC OT Position For Review and Acceptance of Spent Fuel Storage and Handling Applications" and Section 3.8.4 of the USNRC Standard Review Plan.

3.5.5 Design and Analysis Procedures

For the racks, a nonlinear "stick" model representation using masses, springs, dampers and gap elements was used to evaluate stresses and deflections. Acceptable methods were used to account for the effects of sloshing of water and potential sliding of the racks. A three-dimensional finite element model of the pool structure was also constructed in order to investigate the pool structure response. Loads and load combinations were as previously indicated. The pool walls will not be used to provide lateral restraint for the new racks.

A separate analysis of the existing pool slab, the new added column, the existing foundation slab (pile cap), and existing pilings, under the new loadings, was conducted. Acceptable stress levels were found.

Peak responses from accelerations in three directions were combined by the S.R.S.S. method.

A stuck-fuel uplift analysis as well as a dropped fuel bundle analysis was performed, both with satisfactory results.

3.5.6 Acceptance Criteria

Stresses derived from the analysis of the racks were compared with the ASME Boiler and Pressure Code, Section NF. Stress levels in the pool structure and supports were compared with the applicable portions of either the ACI-318-63 Code and American Institute of Steel Construction (AISC) Code.

3.5.7 Materials

Materials used for the racks are in conformance with specifications listed in the ASME B&PV Code and are therefore acceptable.

3.5.8 Summary

The licensee's proposal satisfies the requirements of 10 CFR 50, Appendix A, General Design Criteria 2, 4, 61, and 62 as they apply to structures. Consequently, we conclude that the proposed modification is structurally acceptable.

3.6 Materials Evaluation

The spent fuel racks in the proposed expansion will 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 is composed of boron carbide powder in a rubber-like silicone polymeric matrix. The Boraflex sheets will have a minimum B^{10} content of 0.02 gm/cm^2. The spent fuel storage rack configuration is composed of individual storage cells interconnected to form an integral structure. The

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

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.

The pool contains oxygen-saturated demineralized water containing boric acid, controlled to a temperature below 150°F.

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} inches 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.[2] 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, exposing Boraflex to 1.03 x 10" 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.

[2] J. S. Anderson, "Irradiation Study of Boraflex Neutron Shielding Materials," Brand Industries, Inc., Report 748-10-1, (July 1979).

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. 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.

Tests[1] have shown that neither irradiation, environment nor boron 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.

From our evaluation as disscussed above we conclude that the corrosion that will occur in the H. B. Robinson 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.

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[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).

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

We conclude that the selection of appropriate materials of construction by the licensee meets the requirements of 10 CFR Part 50, Appendix A, Criterion 62, preventing criticality by maintaining structural integrity of components and of the boron poison.

3.7 Occupational Radiation Exposure

We have reviewed the licensee's plan for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. The occupational exposure for this operation is estimated by the licensee to be approximately 173 man-rem. This estimate is based on the licensee's detailed breakdown of occupational exposure for each phase of the modification. The licensee considered the number of individuals performing a specific job, their occupancy time while performing this job, and the average dose rate in the area where the job was being performed. In several instances he is conservative in his estimation of dose-rate and man-hours to perform a specific operation. "Crud" may be released to the pool water because of fuel movements during the proposed SFP modification. This could increase radiation levels in the vicinity of the pool and decrease the clarity of the water. There will be a number of fuel movements in the pool during the modification. Based on experience from prior fuel movements, the plant has not observed significant releases of "crud" to the pool water during refuelings when the spent fuel is first moved into the pool and the addition of "crud" to the pool water is the greatest. The licensee does not expect to have significant releases of "crud" to the pool water during the modification of the pool. The purification system for the pool, which has kept radiation levels in the vicinity of the pool to low levels, includes a filter to remove particles which fall to the floor. The staff concludes that the SFP modification can be performed in a manner that will ensure as low as is reasonably achievable (ALARA) exposures to occupational workers.

The licensee has presented alternative plans for the disposal of the old racks which considered removing and crating intact racks versus removing, cutting and then crating the racks. The licensee is considering three methods of disposal of the old racks: (1) crating the racks whole following surface decontamination for shipment to the burial site; (2) cutting or crushing the old racks into small sections to significantly reduce the volume to be shipped to the burial site; or (3) cutting and electropolishing the racks to remove all radioactive material, then scrapping the decontaminated racks. Cutting the old racks into small sections will permit more efficient packaging in the shipping containers. This will result in a smaller volume of radioactive waste to be disposed of with resulting economic and environmental benefits, e.g., fewer waste shipments and conservation of low waste burial site space. This will also require that the licensee expend effort to cut the old racks and will result in an increase in occupational exposure. At the present time, the licensee has estimated the exposure associated with the disposal of the old racks at a maximum of 2 person-rems, and is performing dose rate surveys on empty spent fuel racks in the spent fuel pool to provide precise data. At this time, taking in account alternative technology, economics and exposures, the licensee will make the final decision as to the choice of method of disassembly and disposal of the old racks so that exposures will be kept to levels that are as low as is reasonably achievable (ALARA).

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies at the unit on the basis of information supplied by the licensee and by utilizing relevant assumptions for occupancy times and for dose rates in the spent fuel area from radionuclide concentrations in the SFP water. The spent fuel assemblies themselves contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the proposed action represents an acceptable impact. 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 exposure burden at the unit. 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.

We conclude that storing additional fuel in the pool will not result in any significant increase in doses received by occupational workers.

3.8 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 May 1970. There will be no change in the waste treatment system or in the conclusions given in Section 2.6 of the evaluation of these systems because of the proposed modification, and therefore, the H. B. Robinson, Unit 2, spent fuel pool expansion is acceptable.

4.0 CONCLUSIONS

On the basis of the foregoing analysis, it is concluded that there will be no significant environmental impact attributable to the proposed action. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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