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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.104 TO FACILITY OPERATING LICENSE NO.DPR-25

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated April 25, 1986 Vermont Yankee Nuclear Power Corporation (VYNPC), the licensee, requested a change to Section 5.5.D of the Technical Specifications for Vermont Yankee Nuclear Power Station (VY). This change would increase the number of fuel assemblies which could be stored in the spent fuel pool from 2,000 to 2870. Other previously approved specifications of Section 5.5 would remain unchanged. The change is based on the installation of new fuel racks in the spent fuel pool which can provide a closer packing of fuel assemblies. Required criticality margins are maintained by incorporation of boron containing material in the rack design. This is a commonly used feature for high density rack design design, and a large number of similar designs have been approved by the NRC.

At this time, the staff is granting the proposed amendment in part: i.e., installation of sufficient fuel storage racks of new design in the pool to accommodate 2870 assemblies, and storage of fuel assemblies in the new racks up to the present Technical Specification limit of 2000 assemblies in the pool. Use of the remaining 870 storage positions for the storage of fuel assemblies is not authorized by this amendment.

2.0 BACKGROUND

VY is a General Electric Company Boiling Water Reactor (BWR) which received an operating license on March 21, 1972. At the time of licensing, the spent fuel pool contained sufficient storage locations to accommodate 600 fuel assemblies. The spent fuel pool cooling system consists of two redundant trains with each train consisting of one 450 gpm pump and one heat exchanger. The design capability of each heat exchanger is 2.23 MBtu/Hr with a pool water temperature of 125°. The spent fuel pool cooling system is non-seismic Category I, non-Class IE.

VYNPC received approval to replace the original spent fuel storage racks with high-density spent fuel storage racks in September 1977. These high-density racks were to be installed in phases providing a total maximum storage capacity of 2000 fuel assemblies. To date, the licensee has installed racks sufficient to store 1690 fuel assemblies. On April 25, 1986 the licensee requested approval to rerack the spent fuel pool for a second time. This second rerack application is the subject of this safety evaluation report. The new high density storage racks would increase the storage capacity of the spent fuel pool to 2870 fuel assemblies and is projected to provide storage capacity until 2001.

The licensee provided additional information on the proposed second rerack request in submittals dated August 15, September 26, October 21 and November 24, 1986, and February 25, March 19, March 31, April 9, April 13, May 22, June 11, September 1, and December 11, 1987, and March 2, 1988. The licensee also incorporated by reference information contained in submittals dated September 11, 1981, November 30, 1983, and May 21, June 27, and December 18, 1984. Information related to the licensee's computer modeling of spent fuel pool cooling was provided at a meeting on January 15, 1987 in Richland, Washington.

In the April 25, 1986 submittal, in addition to requesting approval to rerack the spent fuel pool, the licensee identified necessary changes involving removal of the spent fuel pool cooling system return line spargers and related piping inside the spent fuel pool. In a submittal dated September 1, 1987, the licensee further defined this request by proposing to cut off the Spent Fuel Pool Cooling System (SFPCS) return line at approximately 15 feet above the top of the racks (which is approximately 8 feet below the fuel pool water level). This modification would provide for the storage of an additional 100 fuel assemblies. The licensee stated that the natural circulation developed by the heat generated by the spent fuel will provide adequate cooling for the spent fuel.

The staff issued a status report dated January 21, 1988, which discussed five technical open issues related to the licensee's request to increase the storage capacity of the spent fuel pool to 2870 fuel assemblies. Some of these open issues involved the fuel pool cooling system and its cooling capacity. These open issues also involved increased heat load due to an increase from the present 2000 fuel assemblies limit to the requested 2870 limit. The staff met with the licensee on February 9, 1988 to discuss these issues.

During the meeting, the licensee revealed that it had reached a decision to design, build, and install an enhanced cooling system for the spent fuel pool. This modification was proposed for the purpose of expediting resolution of outstanding issues. Subsequently, the licensee in a submittal dated March 2, 1988, documented its commitment to install an enhanced cooling system. Although no details of the modified design were provided, the licensee did provide some design and performance information for the enhanced SFPCS.

In order to allow reracking to commence in such a way that personnel radiation exposure is minimized, without awaiting completion of review with respect to enhanced cooling, at this time the staff is considering the portion of the proposed expansion involving reracking and placement of the new racks in the pool but is not considering the storage of more than 2000 assemblies in the pool. Consideration of storage of more than 2000 assemblies will await a determination of the adequacy of spent fuel pool cooling for more than 2000 assemblies, including the yet-to-be-designed enhanced spent fuel pool cooling system. Before completing review of the proposed expansion the staff requires more information than is presently available related to the enhanced spent fuel pool cooling system.

Sufficient information is presently available, however, to enable the staff to consider whether or not it is safe to store spent fuel in the new racks up to the present Technical Specification limit of 2000 fuel assemblies, and whether the additional new racks for future storage safely installed in the fuel pool. Because of the pro expansion must be accomplished, it is advantageous to consider the process before sufficient information is available to reach a conclusion wi respect to storing 2870 assemblies.

In order to begin the proposed expansion, the licensee must place a new rack in the pool and transfer fuel presently stored in an old rack to the new rack. The empty old rack is then removed to make room for another new rack, and the process is repeated until all fuel has been transferred to new racks. Additional new racks will than be added to provide space for future storage. It is expected that several months will be required to complete this task for the inventory of irradiated fuel presently in the Vermont Yankee storage pool. If more irradiated fuel were added to the present inventory stored in old racks, the reracking would take even longer and require even more personnel radiation exposure than is required presently.

3.0 EVALUATION

3.1 Criticality Consideration

Required criticality margins are maintained by incorporation of boron containing material in the rack design.

The rack design (described in detail in VYNPC's letter of September 25, 1986) is configured so that the boron associated with the cells, in the form of Boral, is arranged such that there is boron between each pair of fuel assemblies. This includes the Boral on the outer edge of racks, which is arranged so that there is boron between assemblies facing each other across rack gaps. The B-10 loading of the Boral is 0.027 gm/cm^2 minimum. The cell pitch is 6.218 inches and the cell inside width is 5.922 or 6.092 inches (fuel assembly with channel is 5.438 inches).

...lculations for the new racks were performed by Yankee the additional new racks for future storage of 2870 fuel assemblies may be safely installed in the fuel pool. Because of the procedure by which the expansion must be accomplished, it is advantageous to consider the reracking process before sufficient information is available to reach a conclusion with respect to storing 2870 assemblies.

In order to begin the proposed expansion, the licensee must place a new rack in the pool and transfer fuel presently stored in an old rack to the new rack. The empty old rack is then removed to make room for another new rack, and the process is repeated until all fuel has been transferred to new racks. Additional new racks will than be added to provide space for future storage. It is expected that several months will be required to complete this task for the inventory of irradiated fuel presently in the Vermont Yankee storage pool. If more irradiated fuel were added to the present inventory stored in old racks, the reracking would take even longer and require even more personnel radiation exposure than is required presently.

3.0 EVALUATION

3.1 Criticality Consideration

Required criticality margins are maintained by incorporation of boron containing material in the rack design.

The rack design (described in detail in VYNPC's letter of September 25, 1986) is configured so that the boron associated with the cells, in the form of Boral, is arranged such that there is boron between each pair of fuel assemblies. This includes the Boral on the outer edge of racks, which is arranged so that there is boron between assemblies facing each other across rack gaps. The B-10 loading of the Boral is 0.027 gm/cm^2 minimum. The cell pitch is 6.218 inches and the cell inside width is 5.922 or 6.092 inches (fuel assembly with channel is 5.438 inches).

The criticality calculations for the new racks were performed by Yankee

Atomic Electric Company (YAEC). The calculations were performed with two methodologies. The reference criticality analyses were performed with the Monte Carlo code KENO-IV using the NITAWL code to provide cross sections based on the XSDRN code cross section library. For sensitivity calculations and trend analyses the diffusion code PDQ-7 was used with cross sections from the CASMO code. All of these codes and cross sections are well known industry standards, frequently used for analyses of fuel pools and other complex criticality problems, and have been approved by the NRC.

YAEC has benchmarked its KENO methodology against a number of relevant critical experiment results from Babcock and Wilcox and Battelle Northwest Laboratories. These experiments present geometrically representative configurations for fuel racks. YAEC has used these benchmark calculations to develop an analysis methodology uncertainty factor to be added to rack k_{eff} calculations.

YAEC has also determined the potential variation of the rack and fuel parameters that are used in determining the k_{eff} of the rack-fuel system. These parameters include poison thickness, boron concentration, cell pitch, stainless steel thickness and eccentric fuel position. The variation of k_{eff} with these parameters (taken at a 95/95 probability/confidence level) was determined. These (independent parameters) were statistically combined to provide a $\Delta\kappa$ uncertainty which, along with the Monte Carlo statistical uncertainty, is added to the base k_{eff} calculation.

YAEC has investigated abnormal conditions that might be associated with the spent fuel pool and has determined that potential reactivity variations caused by abnormal pool conditions and accidents have either negligible or negative effects on k_{eff}. These include changes in pool temperature from the base conditions, cell or rack displacement from seismic incidents, fuel or heavy object drop events, and fuel assembly placement outside of the racks. Thus k_{eff} for the fuel pool is determined, both for normal and abnormal conditions by adding the previously discussed method and mechanical uncertainties to the base calculation, without the need for additional factors to account for abnormal conditions.

For the base case, the YEAC Monte Carlo calculations assume (1) an infinite square array of cells (2) with a pitch of 6.218 inches, each containing (3) an unirradiated fuel assembly of 64 fuel rods (no water rods) with (4) a uniform enrichment of 3.25 weight percent U-235, (5) no burnable poison and (6) infinite length. The water temperature is 68° F. This fuel assembly enrichment bounds present fuel enrichments and the use of no burnable poison provides conservatism for reactivity calculations.

For the base configuration, the k_{eff} was calculated to be 0.9046. The total uncertainty at a 95/95 level was 0.0221 $\Delta\kappa$, giving a total k_{eff} of 0.9267. This is to be compared to a required upper limit of 0.95.

The fuel assembly lattice used for the base rack calculations was calculated to have a standard reactor core geometry uncontrolled $k\infty$ value of 1.35. YAEC proposed, in the initial submittal, to use a fuel assembly $k\infty$ of 1.35 as the design bases for fuel acceptable for storage in the racks (rather than a fuel enrichment limit). This is common practice for BWR fuel storage (see for example NEDE-24011-P-A-8, May 1986) and allows credit for the burnable poison in the fuel assembly in the analyses to meet the Technical Specification requirement of 0.95. As a result of discussions with the staff concerning the nature of additional uncertainties involved when using a $k\infty$ design criteria, this proposed limit was reduced to 1.31 by VYNPC by letter dated October 21, 1986. The possible reactivity effects of (1) nonuniform enrichment variation in the assembly, (2) uncertainty in the calculation of $k\infty$ and (3) uncertainty in average assembly enrichment were examined and quantified by YAEC, providing the additional correction factor of 0.04 $\Delta\kappa$.

The basic criticality design of the new racks, using boron lined cells to provide the appropriate neutron multiplication level for the closer packed array of high density racks, is a commonly used concept and has been accepted for many spent fuel storage pools. It is an acceptable design concept for maintaining criticality levels for the VY pool.

The methodology used by YAEC to analyze the criticality and reactivity change characteristics of the racks is a state of the art methodology, commonly used and approved for other utilities for such analyses. The Monte Carlo method using the KENO/NITAWL/XSDRN package provides an acceptable methodology for the base calculations and the PDQ/CASMO is acceptable for sensitivity calculations. These methods have been benchmarked against an appropriate selection of critical experiments, with results falling within expected ranges of deviations from the experiments. The derivation of the uncertainty of the methodology from this benchmarking follows normal procedures and also falls within an expected range. It is acceptable.

The examination of uncertainties to be attributed to variances in dimensions and materials in the fuel and racks has covered an acceptable range of parameters and has used a suitable, standard methodology for determining the reactivity effects and their statistical combination. The examination of the effects of abnormal conditions has covered the standard events relating to changes in temperature, movements, misloadings and dropping of assemblies and other equipment, and the results, giving nonpositive reactivity additions, are reasonable and acceptable.

The base calculations and added factors for uncertainties, giving a total k_{eff} of 0.9267, are thus acceptable for an average enrichment of 3.25 percent. There is a margin of 2.3 percent $\Delta\kappa$ to the staff required Technical Specification limit of 0.95. The transfer to a fuel assembly $k\infty$ design basis criterion has conservatively considered relevant additional uncertainty factors, and the resulting design basis $k\infty$ value of 1.31 is acceptable. The approach of using a $\kappa\infty$ design basis has been approved in other applications, and is used in the staff approved General Electric reload analysis approach (as given in GESTAR II, NEDE-24011-P-A-8, May 1986).

The base k_{eff} criterion of 0.95 given in Technical Specification 5.5.B remains the same. Also unchanged by this request is the average enrichment limit of 16 grams of U-325 per longitudinal centimeter of assembly. This specification is compatible with the 3.25 percent U-235 enrichment used in the base calculations.

Therefore, it is concluded that the required criticality margins are maintained by the new racks.

3.2 Structural Engineering

The new high density racks are stainless steel "egg-crate" cellular structures of approximately 6 inches square. Each cell is designed to contain a spent fuel assembly and a typical rack consists of approximately 300 cells whose dimensions are approximately 10 feet long by 8 feet wide and 15 feet high. Weight of the rack and fuel is transmitted to the floor of the pool through supporting legs. The racks are each free-standing on the pool floor and a gap is provided between the racks and between the racks and the pool wall so as to preclude impact during earthquake. Such design provides a margin of safety against tilting and deflection movement.

The spent fuel pool is a reinforced concrete structure supported by the Reactor Building walls. The pool is approximately 26 feet wide by 40 feet long by 39 feet high and is completely lined with seam welded ASTM A240 Type 305 stainless steel.

The licensee's load combinations and acceptance criteria were found to be consistent with those in the "Staff 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 appropriate industry codes such as ASME Section III and ACI 349-80 and the NRC staff guidelines and documents such as Standard Review Plan (NUREG-0800) and Regulatory Guide 1.92 "Combining Model Responses and Spatial Components in Seismic Response Analysis."

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. The hydrodynamic loads of pool water acting on pool walls are considered. 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.

The dynamic response and internal stresses and loads of the racks and pool structure are obtained from a time history seismic analysis. Nonlinear time history analysis are performed utilizing the widely-used industry ANSYS code. Friction between rack support pads and pool floor and hydrodynamic coupling are considered in the analysis. 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 by the licensee to assess the potential effects of a dropped fuel assembly on the racks. The external kinetic energy will be absorbed by rack strain energy through deformation of the rack cells. The overall integrity of the rack will not be adversely affected.

The existing structures were analyzed by the licensee for the modified fuel rack loads. The strength design method for reinforced concrete was used in conjunction with conventional structural analysis procedures to determine capacities. The existing spent fuel pools are determined to safely support the loads generated by the new fuel racks.

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

3.3 Compatability and Chemical Stability of Rack Materials

The staff reviewed the compatibility and chemical stability of the new rack materials wetted by the pool water. The licensee supplemented the original submittal dated April 25, 1986, with additional information regarding rack materials by letter dated March 31, 1987. The proposed spent fuel racks are to be constructed entirely of Type 304L stainless steel, except for threaded rods attached to leveling pads which are 17-4 PH-hardened stainless steel and the neutron absorber material. The 17-4 PH threaded rods are heat treated, chemically cleaned and chrome plated. The neutron absorber material is Boral with a minimum B10 loading of 0.027

gms/cm^2. Boral is a dispersion of boron carbide in an aluminum matrix with an aluminum clad.

The spent fuel rack compartments containing the Boral are not watertight. This will allow venting of gas generated by radiolysis of contained water and by Boral off-gassing, preventing pressure buildup and possible swelling.

The austenitic stainless steel (304L) used in the rack fabrication has a maximum carbon content of 0.03% by weight which minimizes the sensitization in weld heat-affected zones. The stainless steel racks are compatible with the spent fuel pool water that is processed by filtration and demineralization to maintain water purity and clarity. The spent fuel pool purity is maintained at $< 1\mu$ s/cm conductivity at 25°C, < 500 ppb chloride, < 100 ppb total heavy elements, and a pH range of 5.8 to 8.0. Intergranular corrosion tests performed in accordance with ASTM A262, Practice E are required for the austenitic stainless steel. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel rack assemblies, aluminum in Boral neutron absorption plates and zircaloy in the fuel assemblies will not be significant because the materials are protected by highly passiviting oxide films and are, therefore, at similar galvanic potentials.

Boral 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. Boral has been qualified for 10^{11} rads of gamma radiation while maintaining its neutron attenuation capability.

The annulus space in each cell assembly which contains the Boral is vented to the pool to allow venting of radiologic gases and Boral outgassing. This will prevent swelling and bulging of the stainless steel plates.

Tests have shown that Boral does not possess leachable halogens that could be released into the pool environment in the presence of radiation. Similar conclusions have been made regarding the leaching of elemental boron from the Boral.

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 poison coupon surveillance program. Surveillance samples in the form of stainless steel retained sheets of Boral (prototypical of the fuel storage cell walls) will be exposed to the spent fuel pool water. These coupons will be removed and examined periodically over the expected service life.

The staff has reviewed the description of the proposed surveillance program for monitoring the Boral in the spent fuel storage pools and concludes that the program can reveal deterioration that might lead to loss of neutron absorbing capability during the life of the spent fuel racks. The staff does not anticipate that such deterioration will occur, but if it would occur, it would be gradual. In the unlikely event of Boral deterioration in the pool environment, the monitoring program will detect such deterioration and the licensee will have sufficient time to take corrective action, for example, replacement of the Boral sheets.

Based on the above discussion, the staff concludes that the corrosion of the spent fuel pool components due to the spent fuel storage pool environment should be of little significance during the life of the facility. Components in the spent fuel storage pool are constructed of alloys that 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 water indicate that the Boral material will not undergo significant degradation during the projected service life of approximately 40 years for the racks.

The staff further concludes that the environmental compatibility and stability of the materials used in the spent fuel storage pool is adequate based on the test data cited above and actual service experience at operating reactor facilities.

Finally, the staff finds that implementation of the proposed monitoring program and the selection of appropriate materials of construction by the licensee meet the requirements of 10 CFR 50 Appendix A, General Design Criterion 61, regarding the capability to permit appropriate periodic inspection and testing of components and General Design Criterion 62 regarding preventing criticality by maintaining structural integrity of components and of boron poison, and are, therefore, acceptable.

3.4 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for expansion of the spent fuel pool storage capacity with respect to occupational radiation exposure and finds that the ALARA policy, design, and operational considerations are acceptable. This finding is based on the licensee having considered the provisions of 10 CFR Part 20.101, 20.1(c) and 20.103, and the guidelines of Regulatory Guides 8.8 and 8.10 with respect to the planned expansion. The licensee is currently developing specific work packages for the rerack project and has set a dose goal of 20 man-rem. This 20 man-rem dose goal is based on information gained by reviewing the experience at other operating nuclear plants that have recently performed similar spent fuel pool modifications. To meet their dose goal (20 man-rem) and the ALARA provision of 10 CFR 20.1(c), the licensee has assigned an engineer ALARA responsibility to review and approve each work package for the project. The 20 man-rem dose goal represents about 2% of the average annual occupational exposure at the Vermont Yankee plant. The 20 man-rem dose goal includes all activities necessary for the reracking operation including vacuum cleaning of the SFP walls and floor; shuffling fuel, installation of the new racks; removal of the old racks; cleaning decontamination, and any necessary cutting of old racks; and disposal of waste resulting from the reracking operation, including the old racks.

In terms of radiation dose to workers, 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. However, one potential source of radiation to workers during the rerack operation is radioactive activation or corrosion products, which are referred to as crud. Crud may be released to the pool water because of fuel movement during the proposed SFP modification. This could increase radiation levels in the vicinity of the pool. The addition of crud to the pool water is greater during refuelings, when the spent fuel is moved first into the fuel pool. It is at this time that most of the additional crud is introduced into the pool water from the fuel assembly and from the introduction of primary coolant. However, significant releases of crud to the pool water during the rerack operation is not expected, since the new racks are cleaned prior to installation. In addition, the purification system for the pool, which keeps radiation levels in the vicinity of the pool at low levels, includes a filter to remove crud. This filter will be operating during the modification of the pool.

By letter dated November 24, 1986, the licensee provided information describing actions to be taken during spent fuel pool (SFP) modification. Some of the ALARA activities for reducing the occupational radiation dose include:

(a) vacuum cleaning of the SFP floor and walls as required;

(b) hydrolasing and cleaning of old spent fuel racks;

(c) use of remote operations for rack removal and replacement operations; and,

(d) utilizing the SFP Filtration System to maintain clean water in the pool

The licensee also has provided a description of contained and airborne radioactivity sources related to the SFP water which may become airborne as a result of failed fuel and evaporation. The staff has reviewed these source terms and finds them to be acceptable.

Recently there has been a concern expressed that a severe reactor accident could lead to loss of water from the spent fuel pool. Specifically, if the pool cooling system was disabled as part of the reactor accident sequence, and repairs of this system were precluded for several weeks, due to high radiation fields around the plant, then it is possible to postulate a reduction in SFP water inventory. Vermont Yankee, as well as other nuclear plants, employ a defense in depth concept for early warning of, and subsequent protective actions in response to, any accident or abnormal occurrence, including a loss of cooling to the spent fuel pool.

Early warning via monitoring systems and precautions called for by the plant's health physics program assure minimum radiation dose to workers during both normal and abnormal conditions. The spent fuel pool has temperature indicators, water level indicators, vent radiation monitors, an airborne radioactivity monitoring system and an area radiation monitoring system. The water temperature and level indicator provide redundant and diverse means of detecting loss of cooling to the spent fuel pool even during an accident. They provide an early warning, so that corrective actions can be made to restore cooling or to add water before the water level in the spent fuel pool decreases due to boiling.

In addition to the monitoring system and the plant's overall health physics program, the effects of any accident or abnormal condition on personnel including spent fuel pool boiling can be mitigated by implementation of the licensee's emergency plan, which contains re-entry criteria for entering potentially high radiation areas.

On the basis of the above, the staff finds that the projected activities and the dose goal of 20 person-rem for the proposed spent fuel pool expansion is reasonable. Further, we find that the licensee intends to take ALARA considerations into account, to implement reasonable dose reducing activities. Hence, the licensee will be able to maintain individual occupational radiation exposures within the applicable limits of 10 CFR Part 20, and meet the guidelines of Regulatory Guide 8.8. The staff, therefore, finds that the occupational radiation protection aspect of the spent fuel pool modification program is acceptable.

3.5 Radioactive Wastes

The plant contains radioactive waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. The radioactive waste treatment systems have been previously evaluated by the staff and found acceptable. There will be no change in the radioactive waste treatment systems or in the conclusions given regarding the evaluation of these systems as a result of the proposed installation of the new racks. Our evaluation of the radiological considerations supports the conclusions that the proposed installation of new spent fuel storage racks at Vermont Yankee is acceptable. The basis for our conclusions is that the previous evaluation of the radioactive waste treatment systems are unchanged by the installation of new spent fuel storage racks.

The present spent fuel racks will be removed from the SFP and will probably be disposed of as low level waste. If the existing racks are disposed of as solid waste, the volume will be approximately 2000 cubic feet. The annual average volume of solid wastes shipped offsite for burial from Vermont Yankee has been approximately 400 cubic meters. Averaged over the lifetime of the plant the addition of the existing spent fuel racks will increase the total waste volume shipped from the facility by less than 0.4%. This would not have any significant additional environmental impact over that contemplated and discussed in the FES for the operating license application (U.S. Atomic Energy Commission, Environmental Statement Related to the Operation of Vermont Yankee Nuclear Power Station, July 1972).

3.6 Load Handling

3.6.1 Light Loads

A light load is a load that weighs less than the combined weight of a fuel bundle, channel and its handling tool. Since there are no restrictions on the handling of light loads over the spent fuel, a light load could be carried which, if dropped, could have sufficient kinetic energy and impact force on the fuel or rack to potentially result in greater damage than assumed in a fuel handling accident.

In the licensee's September 1, 1987 submittal, the licensee stated that an analysis of light loads normally carried over the spent fuel was performed. The licensee identified a light load to be any load which weighs 700 lbs or less. The results of this analysis indicate that the kinetic energy of these loads is less than that of the design basis fuel handling accident and thus the radiological consequences of a light load drop are bounded by the fuel handling accident. The staff, therefore, considers handling of light loads to be acceptable.

3.6.2 Heavy Loads

Spent fuel storage racks weigh more than a fuel assembly, channel and its handling tool. Thus, spent fuel storage racks are considered to be heavy loads. The reactor building crane will be used to move the storage racks within the reactor building and the spent fuel pool. As part of the review of the Vermont Yankee facility for compliance with guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," the staff concluded in the Safety Evaluation Report dated June 27, 1984, that the reactor building crane was single failure proof by meeting the guidelines of NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants".

In the November 24, 1986 submittal, the licensee provided information that showed the movement of spent fuel within the spent fuel pool, the order of rack replacement, and the path of travel for each of the fuel storage racks. The licensee demonstrated that the storage racks will not be carried over spent fuel or over other racks containing spent fuel. In a subsequent submittal dated February 25, 1987, the licensee provided a drawing that showed the heavy load handling boundaries and laydown areas for the storage racks. The licensee demonstrated that to the extent practical, the paths of travel follow the fuel building structural floor members and beams. The licensee also stated that the load paths and laydown areas will be marked with stanchions and ropes prior to performing heavy load lift. Drawings will be provided to the crane operator in the cab and to the tag man directing the lift to assure adherence to the load paths. The licensee committed to have all deviations from the established load paths approved by management personnel prior to being used. The licensee also committed to prepare installation and removal procedures specifically for the reracking of the spent fuel pool, and to provide qualification, training, and testing of crane operators, as described in D.P. 2201, "Reactor Building and Turbine

Building Crane Operator Qualifications." This information has been reviewed by the staff and found to be acceptable.

Two special lifting devices will be used in the reracking, one for the existing PaR racks and one for the new NES racks. By submittal dated May 22, 1987, the licensee provided drawings of the PaR spent fuel rack lifting rig which show redundancy in the lifting rig. The licensee committed to pre-operationally load test the PaR lifting rig to 150% of the empty spent fuel rack weight. By submittal dated April 13, 1987, the licensee provided drawings of the NES spent fuel rack lifting rig, which also show redundancy in the lifting rig. The licensee committed to pre-operationally load test the NES lifting rig to 150% of the empty spent fuel rack weight (equivalent to 27½ tons) or a total load test equal to 30 tons. In the February 25, 1987 submittal, the licensee committed to ensure that the special lifting devices meet the guidelines of ANSI N14.6-1978, and to perform the load tests and subsequent inspections in accordance with ANSI N14.6-1978.

Based on the above review, the staff concludes that heavy loads handling will be performed in accordance with the guidelines of NUREG-0612, and thus the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control", are met as they relate to proper load handlin, to ensure against an unacceptable release of radioactivity, a criticality accident, or the inability to cool the spent fuel in the spent fuel pool due to postulated load drops. The staff has determined that installation of the new high density racks to provide 2870 storage locations in the VY SFP is acceptable.

3.7 Spent Fuel Shipping Cask Drop Accident

In the licensee's response to the staff's request for additional information dated November 24, 1986, it was stated that the Reactor Building Crane is considered to be single failure proof. Also, the cask drop height to the refueling building floor is less than 30 feet (Ref. FSAR Section 12.2). Therefore, in accordance with Standard Review Plan Section 15.7.5, evaluation findings with respect to radiological consequences for a cask drop accident are not needed. The staff concludes that the proposed expansion meets the applicable criteria with respect to the spent fuel cask drop accident analysis.

3.8 Fuel Handling Accident

The staff independently evaluated a postulated fuel handling accident following the guidance of Standard Review Plant Section 15.7.4, "Radiological Consequences of Fuel Handling Accident", and using the assumptions set forth in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

The calculation was performed by using the staff computer code ACTCODE. The staff conservatively assumed a 24 hour shutdown time for the two damaged fuel assemblies. Credit is given to the Standby Gas Treatment System (SGTS) in the reactor building because the system provides safety grade HEPA filters and charcoal absorbers. Credit is also given for the reactor building, since it maintains a slightly negative pressure during the accident. The radioactivity produced by this accident is processed by the SGTS, which has a 95% removal efficiency for radioactive iodines. The resulting radiological doses at the EAB are 2.58-rem for the thyroid, and 0.337 rem for the whole body. Similarly, at the LPZ, the doses are 0.361 rem for the thyroid and 0/047 rem for the whole body. These doses are far below the criteria of 75 rem for the thyroid and 6 rem for the whole body (SRP 15.7.4).

Because VY's control room does not have charcoal and absorber filters, the staff also considered control room doses due to a fuel handling accident involving a radioactivity release. However, since this release is from a 300 foot high stack, and the atmospheric dispersion factors are in the order of 10^{-6} and 10^{-7} $^{s}/n^{3}$, the effective radiological doses to the control room are estimated to be negligible.

The assumptions used for this analysis are listed as in Table 1.

TABLE 1. ASSUMPTIONS USED IN FUEL HANDLING ACCIDENT

Reactor Power Level	1665 MWth
Number of fuel assemblies in core	368
Number of fuel rods damaged	126
Standby Gas Treatment System filter efficiency for elemental and organic iodines	95%
Cooldown time for impacted spent fuel	24 hrs
Effective pool decontamination factor for iodine	100

GAP ACTIVITY:

Iodine	10%
Krypton	30%
Total noble gas other than Krypton	10%

Location	Time Period	X/Q
EAB	0-2 hrs	0.25×10^{-3} s/m^3
LPZ	0-8 hrs	0.35×10^{-4}
	8-24 hrs	0.21×10^{-4}
	24-96 hrs	0.70×10^{-5}
	96-720 hrs	0.15×10^{-5}

The staff concludes that the proposed spent fuel pool expansion meets the applicable criteria with respect to the fuel handling accident analysis and is acceptable.

3.9 Spent Fuel Cooling System Modification

The license proposed to remove the return line spargers and to terminate the return line in a downward pointing direction at approximately 15 feet above the top of the spent fuel pool storage racks (8 feet below the surface of the water). With the spargers installed as originally licensed in 1972, the water from the SFPCS was returned at the bottom of the spent fuel pool below the spent fuel storage racks. The water generally traveled up through the racks as it passed to the far side of the spent fuel pool, thus providing "forced" cooling of the spent fuel. With the proposed removal of the spargers, the water will enter and exit the pool at approximately the same elevation above the spent fuel storage racks. The mechanism for cooling the spent fuel in this configuration relies on natural circulation. The staff performed an independent spent fuel cooling analysis to verify the licensee's claim that removal of the spargers will not affect spent fuel cooling capability.

The results demonstrate that because of adequate mixing in the upper plenum, the relatively open flow area below the fuel, and the 2-inch gaps around the periphery of the racks, adequate spent fuel cooling is provided regardless of the inlet flow orientation, or "loading patterns" of the hot assemblies within the pool. The primary factor controlling pool performance is the total pool heating rate to total pool recirculation flow rate. Additional details of the staff's independent analysis are contained in NUREG/CR-5048, "Review of the Natural Circulation Effect in the Vermont Yankee Spent-Fuel Pool," by C. L. Wheeler of Pacific Northwest Laboratory.

3.10 Spent Fuel Pool Temperature Limit

Even though this amendment does not modify the current SFP temperature limit, and does not authorize an increase in storage and thus does not affect heat load, the staff addressed the spent fuel temperature limit in its review. Standard Review Plan Section 9.1.3 identifies an acceptable spent fuel pool temperature limit of 140°F for the normal maximum heat load case. Vermont Yankee was originally licensed with Technical

Specification 3.12(H), which limits the maximum pool temperature to 150°F. The licensee stated in the submittal dated April 9, 1987, that the SFPCS is qualified for a pool water temperature of 150°F. Specifically, the qualification temperatures for the major components are: 140°F for the demineralizers, 150°F for the SFPC pumps and heat exchangers, and 175°F for the SFPCS piping. At water temperatures greater than 140°F, the demineralizers resins may start to degrade. In order to prevent degradation of the demineralizer resin, and to be in conformance with the guidelines of SRP Section 9.1.3, the licensee committed in a submittal dated June 11, 1987 to isolate the demineralizers when the SFPCS inlet temperature is 140°F or higher. As detailed in Vermont Yankee's letter of September 1, 1987, spent fuel temperature is continuously monitored when the system is in operation. A Control Room alarm will sound when temperature exceeds an administrati , limit of 125°F. Additionally, Vermont Yankee has committed to directly monitor fuel pool temperature every four hours if one or both fuel pool cooling trains are inoperable (see Vermont Yankee letter, dated September 1, 1987, spent fuel temperature is continuously monitored when the system is in operation. A Control Room alarm will sound when temperature exceeds an administrative limit of 125°. Additionally, Vermont Yankee has committed to directly monitor fuel pool temperature every four hours if one or both fuel pool cooling trains are inoperable (see Vermont Yankee letter, dated September 1, 1987, Attachment 2) until the enhanced Fuel Pool Cooling System is operable.

Further, the licensee performed a re-evaluation of the remaining SFPCS components and determined that each of the components (pump, valves, heat exchangers, etc.), piping and supports, and structures required are capable of operation at a fluid temperature of 200°F. The FSAR states that one purpose of the SFPCS is to assure the operability of the Reactor Building Ventilation (HVAC) system. The licensee has re-evaluated the performance of the reactor building HVAC with a pool water temperature of 200°F and concluded that there will be negligible degradation of the reactor building HVAC system. The licensee also evaluated the available NPSH for the SFPCS pumps with a pool water temperature of 212°F and concluded that there is a 20 foot margin above the required NPSH of 25 feet and thus adequate pump operation can be provided at elevated pool water temperature of 200°F.

Based upon the information reviewed as discussed above, including the 125°F alarm, the staff finds the 150°F maximum pool temperature of Technical Specification 3.12H to be acceptable.

4.0 SIGNFICANT HAZARDS CONSIDERATIONS COMMENTS

The licensee's request for amendment was noticed on June 18, 1986 (51FR22226) and again on December 31, 1986 (51FR47324) with respect to no significant hazards consideration determination and opportunity for hearing. On January 25, 1987 The New England Coalition on Nuclear Pollution (NECNP) and the State of Vermont petitioned to intervene and on January 30, 1987 the Commonwealth of Massachusetts petitioned to intervene. Following ruling on contentions by an Atomic Safety and Licensing Board and a subsequent ruling by an Atomic Safety and Licensing Appeal Board, only one contention remains. That contention concerns the single failure proof characteristics of the spent fuel pool cooling system and the residual heat removal system, and thus is unrelated to this licensing action, because this licensing action does not change the heat load on the spent fuel or residual heat removal systems. This amendment approves the placement of new racks in the spent fuel pool and storage of fuel in the racks without exceeding the presently authorized 2000 assemblies in the pool.

The New England Coalition on Nuclear Pollution (NECNP) provided the only public comments taking issue with the technical basis of the Commission's proposed finding of no significant hazards consideration. In its filing dated July 21, 1986, NECNP expressed the belief that the expansion could significantly increase the risk and consequences of an accident due to the vulnerability of the pools (sic) to failure in the event of a containment failure. This action authorizes only the usage of fuel storage racks of new design and not the storage of additional fuel. The structural capability of the pool has been considered and found to be completely adequate for the additional racks, and the use of racks of new design, rather than old design, has been considered and found not to have a significant impact for reasonably foreseeable design basis events. Therefore, there will not be a significant increase in the risk and consequences of an accident due to vulnerability of the pool in the event of containment failure.

NECNP in its filing dated September 19, 1986 expressed concern that the expansion of the fuel pool storage could increase the probability of a zircaloy cladding fire because of the denser packing of the fuel and the suppression of heat transfer by neutron absorbing material, and increase the consequences of a zircaloy cladding fire by the presence of an increased inventory of radioactivity. NECNP attributed the cause of these accidents to either (1) a reactor accident which by some means, such as a hydrogen explosion, caused a loss of pool water, or (2) an accident which, by some means not involving the reactor, caused a loss of pool water. In a filing dated November 19, 1986 NECNP presented information related to Chernobyl purporting to support the previous filings but introducing no new comments. The Staff's response to NECNP's comment is that the action being authorized does not involve storage of additional fuel; therefore, the comment relating increased consequences to increased inventory does not apply. With respect to the remaining concern related to increased probability of an accident because of the new rack design, the staff, in Section 3.8 of this evaluation, has addressed both the safety and environmental aspects of a fuel handling accident, an event which bounds the potential adverse consequences of accidents attributable to operation of a spent fuel pool with high density racks. A fuel handling accident may be viewed as a "reasonably foreseeable" design basis event which the pool and its associated structures, systems and components (including the racks) are designed and constructed to prevent. The environmental impacts of this accident were found not to be significant.

The staff has considered events whose consequences might exceed a fuel handling accident, that is, beyond design basis events. Such occurrences include a criticality accident and a zircaloy cladding fire caused by overheating following the loss of spent fuel pool cooling caused by a pool failure. Compliance with General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control" and 62, "Prevention of Criticality in Fuel Storage and Handling" of 10 CFR Part 50, Appendix A, and adherence to approved industry codes and standards as set forth in the licensee's rerack application (which includes compliance with certain design and construction criteria contained in the Final Safety Analysis Report) provides assurance that such events are of very low probability by ensuring that pool and rack integrity and pool cooling capability are maintained. Acceptance criteria for the General Design Criteria consider all reasonably foreseeable events. For example, in this case, criticality is prevented by providing very strong racks, which will maintain the proper spacing between fuel assemblies; the spent fuel pool walls are made of reinforced concrete four or more feet thick, rendering pool wall failure a very unlikely event.

The environmental impacts of criticality and pool wall failure could be significant; however, neither of these events is considered to be reasonably foreseeable in light of the design of the spent fuel pool and racks. Therefore, further discussion of their impacts is not warranted and the staff concludes that the reasonably foreseeable environmental impacts attributable to the proposed action are not significant.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The licensee's request for amendment to the operating license for Vermont Yankee including a proposed determination by the staff of no significant hazards consideration was individually noticed in the Federal Register on June 18, 1986, followed by a notice on December 31, 1986, pertaining specifically to the hybrid hearing provisions of the Commission's regulations. At this time the staff is considering only the reracking (installation of sufficient fuel storage racks in the pool to accommodate 2870 assemblies, and storage of fuel assemblies in the new racks up to the present Technical Specification limit of 2000 assemblies in the pool).

The Commission's regulations in 10 CFR 50.92 include three standards used by the NRC staff to arrive at a determination that a request for amendment involves no significant hazards considerations. These regulations state that the Commission may make such a final determination if operation of a facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed spent fuel pool expansion amendment is similar to more than 100 earlier requests from other utilities for spent fuel pool expansions. The majority of these requests have already been granted by the NRC; others are under staff review. The knowledge and experience gained by the NRC staff in reviewing and evaluating these similar requests were used in this evaluation. The licensee's request does not use any new or unproven technology in either the analytical techniques necessary to support the expansion or in the construction process.

The staff has determined that reracking the spent fuel pool at the Vermont Yankee Nuclear Power Station does not significantly increase the probability or consequences of accidents previously evaluated; does not create new accidents not previously evaluated, and does not result in any significant reduction in the margins of safety with respect to criticality, cooling or structural considerations.

The following staff evaluation in relation to the three standards demonstrates that the proposed amendment for the SFP expansion does not involve a significant hazards consideration.

First Standard

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

The following postulated accidents and events involving spent fuel storage have been identified and evaluated by the licensee. The staff likewise evaluated the same accidents and events.

1. A spent fuel assembly drop in the spent fuel pool.

2. A seismic event.

3. A spent fuel cask drop.

4. A construction accident.

The probability of occurrence of any of the first three accidents is not affected by the racks themselves; thus the modification cannot increase the probability of occurrence of these accidents. As for the construction accident, the licensee will not carry any rack directly over the stored spent fuel assemblies. All work in the spent fuel pool area will be controlled and performed in strict accordance with specific written procedures. The crane that will be used to move the racks within the reactor building and the spent fuel pool has been evaluated and found acceptable. Section 3.6 of this safety evaluation contains the details of the staff's analysis. Thus, the probability of a construction accident is not significantly increased as a result of reracking. Accordingly, the proposed modification does not involve a significant increase in the probability of occurrence of an accident previously evaluated.

As noted in Section 3.1 of this safety evaluation, the consequences of a spent fuel assembly drop in the spent fuel pool was evaluated and it was found that the criticality acceptance criterion, k_{eff} less than or equal to 0.95, is not violated. The staff also conducted an evaluation of the potential consequences of a fuel handling accident. The staff analysis found that the calculated doses are less than 10 CFR Part 100 guidelines. The results of the analysis show that dropping a spent fuel assembly on the racks will not distort the racks such that they will not perform their safety function. Section 3.6 contains the details of the staff's accident analysis. Thus, the consequences of this type of accident are not significantly changed from the previously evaluated spent fuel assembly drops which have been found acceptable.

The consequences of a seismic event have been evaluated and are acceptable. The new racks will be designed and fabricated to meet the requirements of applicable portions of the NRC Regulatory Guides and published standards. The new free-standing racks are designed, as are the existing free-standing racks, so that the floor loading from racks completely filled with spent fuel assemblies, partially filled, or empty at the time of the incident, does not exceed the structural capability of the spent fuel pool. The Reactor Building and spent fuel pool structure have been evaluated for the increased loading from the spent fuel racks in accordance with the criteria previously evaluated by the staff and found acceptable. Section 3.2 contains the details of the staff's analysis. Thus, the consequences of a seismic event are not significantly increased from previously evaluated events.

The consequences of a spent fuel cask drop have been evaluated (see Section 3.7 of this safety evaluation). Because the Reactor Building Crane is single failure proof and the cask drop height to the refueling floor is less than 30 feet, the radiological consequences of cask drop meets the applicable criteria and are not significantly increased from previous analysis. The consequences of a construction accident are enveloped by the spent fuel cask drop analysis. No rack (old or new) weights more than a single 25 ton cask. In addition, all movements of heavy loads handled during the rerack operation will comply with the NRC guidelines presented in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The consequences of a construction accident are not increased from previously evaluated accident analyses.

Therefore, it is concluded that the proposed amendment to replace the spent fuel racks in the spent fuel pool will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Second Standard

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

As noted in various sections of this safety evaluation, the staff evaluated the proposed modification in accordance with the guidance of appropriate NRC Regulatory Guides, appropriate NRC Standard Review Plans, and appropriate industry codes and standards. In addition, the staff has reviewed several previous NRC Safety Evaluations for rerack applications similar to this proposal. No unproven techniques and methodologies were utilized in the analysis and design of the proposed high density racks. No unproven technology will be utilized in the fabrication and installation process of the new racks. The basic reracking technology in this case has been developed and demonstrated in numerous applications for a fuel pool capacity increase which have already received NRC staff approval.

Therefore it is concluded that the proposed amendment to replace the spent fuel racks in the spent fuel pool will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

"Involve a significant reduction in a margin of safety."

The staff Safety Evaluation review process has established that the issue of margin of safety, when applied to a reracking modification, should address the following areas:

1. Nuclear criticality considerations.

2. Thermal-hydraulic considerations.

3. Mechanical, material and structural considerations.

The established acceptance criterion for criticality is that the neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, including all uncertainties, under all conditions. This margin of safety has been adhered to in the criticality analysis methods for the new rack design.

The methods used in the criticality analysis conform with the applicable portions of the appropriate staff guidance and industry codes, standards, and specifications. In meeting the acceptance criteria for criticality in the spent fuel pool, such that k_{eff} is always less than 0.95, including uncertainties at a 95%/95% probability/confidence level, the proposed amendment to rerack the spent fuel pool does not involve a significant reduction in a margin of safety for nuclear criticality. Section 3.1 contains the details of the staff's analysis.

Reracking the Vermont Yankee spent fuel storage pool, without approving expanded fuel storage capacity, adds nothing to the pool heat load. Therefore, all thermal-hydraulic considerations related to bulk pool temperature and the spent fuel pool cooling system remain unchanged. Local cooling effects due to removal of the return line spargers was independently analyzed by the staff, and found to cause no significant reduction in the margin of safety.

The main safety function of the spent fuel pool and the racks is to maintain the spent fuel assemblies in a safe configuration through all normal or abnormal loadings, such as an earthquake, impact due to a spent fuel cask drop, drop of a spent fuel assembly, or drop of any other heavy object. The mechanical, material, and structural design of the new spent fuel racks is in accordance with applicable portions of the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1987, as modified January 18, 1979; Standard Review Plan 3.8.4; and other applicable NRC guidance and industry codes. The rack materials used are compatible with the spent fuel pool and the spent fuel assemblies (see Section 3.3 of this safety evaluation). The structural considerations of the new racks address margins of safety against tilting and deflection movement, such that the racks are not damaged during impact (see Section 3.2 of this safety evaluation). In addition, the spent fuel assemblies remain intact and no criticality concerns exist. Thus, the margins of safety are not significantly reduced by the proposed rerack.

Therefore, it is concluded that the proposed amendment to replace the spent fuel racks in the spent fuel pool will not involve a significant reduction in a margin of safety.

6.0 SUMMARY

Based on the above-described review, the staff concludes that the reracking of the Vermont Yankee spent fuel pool to accommodate 2870 fuel assemblies using the new high density racks is acceptable. The present Technical Specification Section 5.5, which limits the number of spent fuel assemblies stored in the spent fuel pool to 2000 assemblies, remains unchanged.

The staff's conclusions are limited to the removal of the spargers and the use of the new racks. The staff is not at this time authorizing the filling of the racks beyond the 2000 assemblies presently authorized.

7.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that the proposed amendment involves no significant hazards consideration, and in Section 5.0 of this evaluation the Commission reaches a final conclusion that this amendment involves no significant hazards consideration. Therefore, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

8.0 CONCLUSIONS

The staff has reviewed and evaluated the licensee's request for reracking the Vermont Yankee spent fuel pool. Based on the considerations discussed in this safety evaluation, the staff concludes that:

(1) This amendment will not (a) significantly increase the probability or consequences of accidents previously evaluated, (b) create the possibility of a new or different accident from any accident previously evaluated, or (c) significantly reduce a margin of safety; and therefore, the amendment does not involve significant hazards considerations;

(2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and

(3) 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.

Therefore reracking with the new racks is approved.

Principal Contributors: S. Kim, H. Richings, A. Chu, M. Lamastra, J. Lee, F. Witt, J. Ridgely, and V. Rooney

Date of Issuance: May 20, 1938