UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 182 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated September 4, 1998, as supplemented on February 8, April 16, August 26, September 16, and November 17, 1999, the Vermont Yankee Nuclear Power Corporation (the licensee) submitted a request to amend the Vermont Yankee Nuclear Power Station (Vermont Yankee) Technical Specifications (TSs). The proposed amendment would increase the spent fuel pool (SFP) storage capacity from 2,870 to 3,353 fuel assemblies. The supplemental information did not affect the staff's proposed no significant hazards consideration, and was within the scope of the original amendment application as published.

2.0 EVALUATION

The licensee proposed a change to TS section 5.5, "Spent and New Fuel Storage." The proposed change would change TS 5.5.D to state "The number of spent fuel assemblies stored in the spent fuel pool shall not exceed 3353." The current TS states that the number of spent fuel assemblies stored in the spent fuel pool shall not exceed 2870.

The NRC staff's evaluation of the proposed change is presented below.

2.1 Heavy Loads

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides recommendations and guidelines for licensees to assure safe handling of heavy loads in proximity to or over safe shutdown equipment, irradiated fuel, and the reactor core. The objectives of the guidelines are to assure that either (1) the potential for a load drop is extremely small, or (2) the potential hazards of a load drop do not exceed acceptable limits. NUREG-0612 provides guidelines that are implemented in two phases. Phase I of the guidelines addresses measures for reducing the likelihood of dropping heavy loads in proximity to or over safe shutdown equipment, fuel in the reactor core, and the spent fuel pool. The Phase I guidelines provide criteria for establishing safe load paths, procedures for load handling operations, training of crane operators, and design, testing, inspection, and maintenance of cranes and lifting devices. Phase II of the guidelines addresses alternatives for reducing and mitigating the consequences of heavy load drops, including using (1) a single-failure-proof crane for increased handling system reliability, (2) electrical interlocks and mechanical stops for restricting crane travel, or (3) load drop and consequence analyses for assessing the impact of dropped loads on plant safety and operations.

Generic Letter (GL) 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612," dated June 28, 1985, dismissed the need for licensees to implement the guidelines of NUREG-0612, Phase II. However, GL 85-11 encouraged licensees to implement actions they perceived to be appropriate to maintain safety.

The NRC's letter to the licensee dated June 27, 1984, approved the implementation of NUREG-0612 at Vermont Yankee. Also, in the previously approved reracking requests, the licensee addressed issues pertaining to NUREG-0612. In this request, the licensee addresses the movement of spent fuel assemblies, installation of spent fuel storage racks, the design and uses of the Reactor Building hoisting system, establishment of safe load paths, the use of procedures, crane operator training, and postulated load drop accident analyses and consequences over the spent fuel pool and safety-related equipment.

2.1.1 Hoisting System Evaluation

The reactor building 110-ton overhead bridge crane will be used to install the new racks and handle other heavy loads during the rerack operation. As stated in the NRC's letter to the licensee dated June 27, 1984, the crane is single-failure-proof in accordance with criteria in NUREG-0554. Furthermore, the crane is designed in accordance with requirements of the Electric Overhead Crane Institute (EOCI), Specification No. 61, and the Crane Manufacturers Association of America (CMAA), "Specification No. 70 for Electric Overhead Traveling Cranes," and ANSI B30.2-1976, "Overhead and Gantry Cranes (Top Running Bridge and Multiple Girder)." The rated capacity of the crane is 110 tons in the main hoist, and 7 tons in the auxiliary hoist. The maximum load of the rack and the lifting device is 27,160 pounds.

As stated in the Vermont Yankee Final Safety Analysis Report (FSAR), because of the single-failure-proof capability of the crane, all the components in the load path of the main hoist either are redundant or have a large factor of safety (minimum of 10 for redundant load paths). The redundant load paths for the main hook consist of the attachment point, load block, cable, reversing sheaves, drum, gear drive, brakes, limit switches for overloading and over-hoisting, and an over-speed switch.

A remotely controlled lifting rig will be interposed between the hook of the Reactor Building overhead crane and the spent fuel racks. The licensee states that the Vermont Yankee lifting-rig is specifically designed to lift the new spent fuel rack modules and is similar in design to the rigs used by a number of other plants. The lifting rig is designed and tested in accordance with the guidelines in NUREG-0612 and requirements in ANSI N14.6 (1978), "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials." It consists of four independently loaded lift rods and configured such that failure of a single rod will not result in uncontrolled lowering of the rack. As stated by the licensee, both the stress design and the load testing of the lifting rig satisfy guidelines in Section 5.1.6(1) of NUREG-0612 and ANSI N14.6 (1978), respectively. Accordingly, the lift rods are designed as follows: (1) with a stress design factor (as specified in ANSI N14.6, Section 3.2.1) of three times the combined weight to be lifted plus the weight of the lifting device without exceeding the minimum yield strength of the material; (2) five times the lifted weight without exceeding the ultimate strength of the material; (3) load tested to 300 percent of the maximum weight to be lifted; and (4) after load testing, examination of the critical weld joints using a liquid penetrant or magnetic particle. Non-customized lifting devices (i.e., slings) will be used In accordance with NUREG-0612 and ANSI B30.9-1971, "Slings." Therefore, the slings shall be proof tested at a minimum of 1.5 times their rated capacity in accordance with Section 9.3.3 in ANSI B30.9.

The staff believes that the single-failure-proof crane coupled with the design and testing of the lifting rig and other lifting devices will enable the licensee to handle heavy loads with little to no risks to the safety of the rerack operation.

2.1.2 Analysis of Heavy Load Drop Accidents

Vermont Yankee's FSAR and previous NRC letters to the licensee concluded that due to the single-failure-proof capability of the handling system a load drop is not feasible because the crane would be able to retain the load or transfer the load to another load path. Also, the potential for a rack drop to impact fuel is unlikely because the licensee plans to shuffle any fuel in the load path to racks outside the load path prior to any load movement.

Furthermore, NUREG-0612 recommends that licensees provide an adequate defense-in-depth approach to maintaining safety during the handling of heavy loads near spent fuel and cites four major causes of accidents: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. The licensee states that it will implement measures using administrative controls and procedures to preclude load drop accidents in these four areas. The licensee plans to provide: (1) comprehensive training to the rerack installation crew, (2) use redundantly designed lifting rigs, (3) perform inspection and maintenance checks on the cranes and lifting devices prior to the rerack operation, and (4) use specific procedures that cover the entire rerack effort, including the identification of required equipment, inspection, acceptance criteria prior to load movement, defining safe load paths, and steps and precautions for proper load handling and movement. In addition, the licensee will restrict travel of the racks from over fuel in the SFP and safety-related equipment, and limit the lift heights of the racks to 6 inches above the SFP operating floor.

The NRC staff agrees with the licensee that the use of a single failure proof crane in conjunction with administrative procedures and controls focused on, but not limited to, the areas noted above will enable the licensee to maintain safety during the rerack operation .

2.1.3 Heavy Loads Conclusion

Based on the preceding discussion, we find that the licensee's considerations for the movement of heavy loads to support proposed changes to the CTS 5.5 and the increase in the spent fuel pool storage capacity are acceptable. The licensee's use of the Reactor Building overhead crane, the lifting rig, and the administrative controls and procedures are in accordance with the guidelines in NUREG-0612 and ANSI N14.6. The increased reliability of the crane coupled with the design, testing, and inspection of the lifting rig and other lifting devices will enable the licensee to safely handle the racks and other heavy loads during the rerack operation. The administrative controls and procedures to improve the handling and control of the racks further enhance the licensee's capability to reduce the potential for a load drop.

2.2 Spent Fuel Cooling

2.2.1 Spent Fuel Pool Cooling and Cleaning System

The safety function of the Fuel Pool Cooling and Demineralizer System (FPCDS) is to remove the decay heat generated by stored spent fuel assemblies and maintain pool temperature. The system also maintains a specified fuel pool water purity, clarity, and level. The FPCDS consists of a non-safety-related normal fuel pool cooling subsystem (NFPCS) and a safety-related standby fuel pool cooling subsystem (SFPCS).

Normal spent fuel cooling and pool water cleaning is provided by the NFPCS. The subsystem is rated as nonseismic Category I and includes two pumps, two heat exchangers, and two filter demineralizers, which are in parallel trains. One train is normally aligned and operating during plant operation. Each pump has a design flow rate of 450 gpm. The NFPCS transfers decay heat to the reactor building closed cooling water system.

The SFPCS is utilized for periods when the spent fuel decay heat loads exceed that capacity of the NFPCS or in the event of a design basis accident, such as a seismic event. The SFPCS is a redundant, Seismic Class I, Safety Class 3 system which includes two pumps and two heat exchangers in parallel trains. Both trains are normally lined up in standby mode and can be placed in service remotely from the control room. Each pump has a design flow rate of 700 gpm. The SFPCS is cooled by the safety-related station service water system.

During refueling operations, when the reactor cavity and equipment pit are filled with water and the refueling gates are removed for fuel movement, the residual heat removal (RHR) system may also operate to remove all or a portion of the decay heat from the reactor cavity or spent fuel pool. A single train of the RHR system is available for use.

The pool has an administrative limit of 125 °F. The design-basis operating temperature limit of the pool is 150 °F. The demineralizers are isolated at 140 °F to prevent degradation of the resins. The SFP has a water temperature monitor which alarms in the control room when the SFP water temperature reaches 120 °F. Annunciator response instruction lists the probable causes and corrective actions to be taken when the high temperature alarm is received. Under normal conditions, the temperature is controlled administratively between 80 - 110 °F using NFPCS. If the pool temperature exceeds 120 °F, the temperature is trended, NFPCS is secured, and SFPCS is placed in service prior to exceeding 140 °F.

Heat Removal Capacities

NFPCS	1 train	4.4 MBtu/hr
SFPCS	1 train	10.5 MBtu/hr
	2 trains	21 MBtu/hr
Combined subsystems	1 train NFPCS and 1 train SFPCS	14.9 MBtu/hr
RHR	1 train	41.9 MBtu/hr

As a result of an increase in the spent fuel assemblies planned to be stored in the SFP, the decay heat load for any specific fuel discharge scenario will increase. The licensee performed analysis for the two discharge scenarios to evaluate the effects of increased SFP storage capacity on the FPCDS and to establish the bounding heat load capacity for a planned and unplanned offload.

For the evaluation of spent fuel pool cooling capacity for planned offloads, the licensee must assume a single failure and maintain the pool temperature below 150 °F. Routine refueling outages at Vermont Yankee utilize partial-core offloads. The licensee calculated the bounding heat generation from a partial-core offload to be 10.5 MBtu/hr at 6 days after shutdown. The single active failure was postulated to be a loss of offsite power during shutdown or a single active failure of plant equipment. In the case of loss of offsite power during shutdown, an emergency diesel generator may be out of service for testing or maintenance and, therefore, only one train of NFPCS and one train of SFPCS would be available from the remaining emergency diesel generator. The heat load generated from a partial-core offload is within the heat removal capacity of one train of SFPCS. The staff performed an independent calculation and found a similar value for the decay heat load generated by this scenario. Therefore, we agree that sufficient capacity exists in the FPCDS to maintain the pool temperature below 150 °F assuming a single failure.

Consistent with the standard review plan and recent approvals for spent fuel pool storage expansion, the licensee was requested to evaluate the cooling capacity for the heat load generated from an unplanned offload. For an unplanned offload, one refueling load after 36 days decay and a full-core offload is evaluated. A single failure does not need to be assumed and the pool temperature must be maintained below boiling. Therefore, for Vermont Yankee, all trains of SFPCS and the NFPCS would be available. The licensee approximated the decay heat load to be 25.38 MBtu/hr at 10 days after shutdown. The licensee used a conservative value of 26 MBtu/hr and calculated the equilibrium pool temperature would be approximately 165 °F. The staff performed an independent calculation and found a similar value for the decay heat load generated by this scenario. Therefore, we agree the heat load generated from an unplanned, full-core offload is within the capacity of the combination of the NFPCS and SFPCS to maintain the bulk water below boiling.

In the FSAR the licensee states that fuel movement is based on the availability of each heat removal system, its capacity, and the actual heat load. In the supplemental submittal dated August 26, 1999, the licensee states that the calculations for FPCDS capability, using 6-day and 10-day decay times, assume that the refueling cavity gates are installed. When the gates are removed, shutdown cooling is used during the offload of the reactor vessel in accordance with plant procedures. Shutdown cooling uses one train of RHR and FPCDS to cool the vessel and connected spent fuel pool. One train of the RHR system has a heat removal capacity of 41.9 MBtu/hr, which is adequate to remove the spent fuel heat load when the gates are removed.

Based on the NRC staff's review described in this section and independent calculations, we find that the FPCDS provides the licensee with sufficient heat removal capacity to maintain water temperature below 150 °F for planned, partial-core offloads and below the point of boiling for unplanned offloads.

2.2.2 Effects of SFP Boiling

In the supplemental submittal dated August 26, 1999, the licensee stated that due to the redundancy and diversity of the spent fuel pool cooling systems, a loss of all fuel pool cooling is a beyond design and licensing basis event. Regardless, in the event that there is a complete loss of cooling capability using FPCDS heat exchangers to remove heat from the SFP, the SFP water temperature will begin to rise and eventually will reach the boiling temperature. The planned refueling offload consists of a maximum of 136 fuel assemblies of the 368 fuel assemblies in a full core. The licensee performed an analysis which concludes that the calculated minimum time from the loss-of-pool cooling until the pool boils is 11.12 hours for a planned partial-core offload with a decay heat load of 10.5 MBtu/hr after 6 days of decay time.

In the supplemental submittal dated August 26, 1999, the licensee calculated that the maximum boiloff rate would be 39.38 gpm from the heat load generated from an unplanned, full-core offload. The licensee calculated the boiloff rate from a full-core offload since it bounds a partial-core offload. The licensee has multiple sources of makeup exceeding this boiloff rate, including the condensate transfer system and demineralized water system. Makeup can also be provided by the seismic Category I service water system.

Based on our review, due to the flexibility and diversity in the redundant Vermont Yankee SFP cooling subsystems and because the SFPCS is a seismic Category I system that has a Class 1E power supply, we find that the loss of all spent fuel pool cooling is a beyond design and licensing basis event for Vermont Yankee. Further, we find that the licensee has sufficient time and makeup capacity so that there will not be a loss of inventory due to boiloff from a planned, partial-core offload.

2.2.3 Spent Fuel Pool Cooling Conclusion

Based on the NRC staff's review of the licensee's submittals and our independent calculations, the NRC staff finds the proposed TS change to increase the SFP storage capacity from 2,870 to 3,353 fuel assemblies at Vermont Yankee with respect to the SFP cooling capacities acceptable.

2.3 Structural Integrity

The primary purpose of the NRC staff's review in this area was to assure the structural integrity and functionality of the racks, the stored fuel assemblies and the SFP structure subject to the effects of the postulated loads (Appendix D of Standard Review Plan (SRP) Section 3.8.4) and fuel handling accidents.

2.3.1 Storage Racks

There are nine existing racks installed in the SFP at Vermont Yankee. The total storage capacity allowed by the current TS is 2,870 fuel assemblies. The licensee proposed adding 3 new racks to increase the total storage capacity to 3353 fuel assemblies. All 12 storage racks are seismic Category I equipment and are required to remain functional during and after a safe shutdown earthquake (SSE). The licensee, with its contractor Holtec, performed structural analyses of the racks for the requested license amendment.

The computer program DYNARACK was used for dynamic analysis to demonstrate the structural adequacy of the Vermont Yankee spent fuel rack design under the combined effects of earthquake and other applicable loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment, and they are not attached to the floor or walls of the SFP. A non-linear dynamic model consisting of inertial mass elements, spring elements, gap elements, and friction elements, as defined in the program, were used to simulate the three-dimensional (3-D) dynamic behavior of the rack and the stored fuel assemblies including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

Analyses of two models were performed: a 3-D single rack (SR) model and a 3-D whole pool multi-rack (MR) model. For the 3-D SR analyses several configurations were used. The rack was considered to be fully loaded, half loaded, and almost empty with three different coefficients of friction (μ=0.2, 0.8 and a random value where the mean is about 0.5) between the rack pedestal and the pool floor to investigate the stability of the rack with respect to overturning. For the 3-D MR analyses, 12 free-standing racks were considered fully loaded with three different coefficients of friction to investigate the fluid-structure interaction effects between the racks and the pool walls as well as those among the racks and to identify the worst-case response for rack movement and for rack member stresses.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration components) were generated from the design response spectra defined in the Vermont Yankee FSAR. The licensee demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra as well as matching a target power spectral density (PSD) function compatible with the design response spectra as discussed in SRP Section 3.7.1.

A total of 31 3-D SR and MR analyses were performed. The racks were subjected to the service, upset and faulted loading conditions (Level A, B and D service limits). The results of the analyses show that the maximum displacement of the racks at the top is about 1.48 inches indicating that there is adequate safety margin against overturning of the racks. The results of the analyses also show that there is no impact potential between the racks and between the rack and the pool wall. The calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension were compared with corresponding allowable stresses specified in ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Subsection NF. The results show that all induced stresses under the SSE loading condition are smaller than the corresponding allowable stresses specified in the ASME Boiler and Pressure Vessel Code indicating that the rack design is adequate.

The licensee also calculated the rack weld stresses at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal, and cell-to-cell connections) under the dynamic loading conditions. The licensee demonstrated that all of the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code indicating that the weld connection design of the rack is adequate.

Based on: (1) the licensee's comprehensive parametric study (e.g., varying coefficients of friction and fuel loading conditions of the rack), (2) the adequate factor of safety of the induced stresses in the rack when they are compared to the corresponding allowables provided in the ASME Code, and (3) the licensee's overall structural integrity conclusions supported by both SR and MR analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and, therefore, are acceptable.

2.3.2 Spent Fuel Storage Pool

The licensee analyzed the SFP to demonstrate the adequacy of the structures under fully loaded fuel racks with all storage locations occupied by fuel assemblies. The fully loaded structures were subjected to the load combinations specified in the Vermont Yankee FSAR.

In the licensee's response of September 16, 1999, to the staff's request for additional information, the licensee demonstrated that the predicted factors of safety varied from 1.07 to 1.46 for shear force and bending moment of the concrete walls and slab. In view of the calculated factors of safety, the staff concludes that the licensee's structural analyses demonstrate the adequacy and integrity of the structures under full-fuel loading, thermal loading and SSE loading conditions. Thus, the storage fuel pool design is acceptable.

2.3.3 Structural Integrity Associated with a Postulated Fuel Handling Accident

The following two refueling accident cases were evaluated by Vermont Yankee: (1) drop of a fuel assembly with its handling tool, which impacts the baseplate (deep-drop scenario) and (2) drop of a fuel assembly with its handling tool, which impacts the top of a rack (shallow drop scenario).

The analysis results of accident case (1) show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads located near the fuel handling area; therefore, the liner would not be ruptured by the impact as a result of the fuel assembly drop through the rack structure. The analysis results of accident drop case (2) show that damage will be restricted to a depth of 7.43 inches below the top of the rack, which is above the active fuel region. The staff reviewed the licensee's analysis and concurs with its findings. This is acceptable based on Vermont Yankee's structural integrity conclusions supported by the parametric studies.

2.3.4 Structural Integrity Conclusion

Based on the NRC staff's review and evaluation of the licensee's submittal and responses to the requests for additional information, the staff concludes that the licensee's structural analysis and design of the spent fuel rack modules and the SFP structures are adequate to withstand the effects of the applicable loads including that of the SSE. The analysis and design are in compliance with the current licensing basis set forth in the FSAR and applicable provisions of the SRP, and are therefore, acceptable.

2.4 Radiological Effects

2.4.1 Occupational Radiation Exposure

The staff reviewed the licensee's plan for installation of additional spent fuel racks at Vermont Yankee with respect to occupational radiation exposure. A number of facilities have performed similar operations in the past. On the basis of the lessons learned from these operations, the licensee estimates that the proposed fuel rack installation can be performed for between 1.6 and 3 person-rem.

All of the operations involved in the fuel rack installation will utilize detailed procedures prepared with full consideration of ALARA (as low as reasonably achievable) principles. The Radiation Protection Department will prepare Radiation Work Permits (RWPs) for the various jobs associated with the SFP rack installation operation. These RWPs will instruct the project personnel in the areas of protective clothing, general dose rates, contamination levels (including potential exposure to hot particles), and dosimetry requirements. Each member of the project team will attend an ALARA Pre-Plan meeting and each team member will be required to attend daily pre-job briefings on the scope of the work to be preformed. Personnel will wear protective clothing and will be required to wear personnel monitoring equipment including alarming dosimeters.

Since this license amendment does not involve the removal of any spent fuel racks, the licensee does not plan on using divers for this project. However, if it becomes necessary to utilize divers to remove any interferences which may impede the installation of the new spent fuel racks, the licensee will equip each diver with radiation detectors with remote, above surface, readouts which will be continuously monitored by Radiation Protection personnel. The licensee will conduct radiation surveys of the diving area prior to each diving operation and following the movement of any irradiated hardware. In order to minimize diver dose, the licensee will use visual barriers (such as streamers fastened to rope, nets, or enclosure) as much as practical. The licensee will monitor and control personnel traffic and equipment movement in the SFP area to minimize contamination and to ensure that exposure is maintained ALARA.

On the basis of the NRC staff's review of the Vermont Yankee proposal, the staff concludes that the Vermont Yankee SFP rack modification can be performed in a manner that will ensure that doses to workers will be maintained ALARA. The staff finds the projected dose for the project of 1.6 to 3 person-rem to be in the range of doses for similar SFP modifications at other plants and is, therefore, acceptable.

2.4.2 Solid Radioactive Waste

Spent resins are generated by the processing of SFP water through the SFP Purification System. The licensee does not expect the resin change-out frequency of the SFP purification system to be permanently increased as a result of the storage of additional spent fuel assemblies in the SFP. In order to maintain the SFP water as clean as possible, and thereby minimize the generation of spent resins, the licensee will vacuum the floor of the SFP to remove any radioactive crud and other debris before the new fuel rack modules are installed. Overall, the licensee does not expect that the additional fuel storage made available by the increased storage capacity will result in a significant change in the generation of solid radioactive waste. The staff finds the licensee's conclusion to be acceptable.

2.4.3 Design Basis Accidents

On April 25, 1986, Vermont Yankee submitted an amendment request to increase the SFP capacity from 2000 to 2870. The staff approved that amendment request on May 20, 1988. The staff's safety evaluation supporting the issuance of that amendment concluded that the licensee's fuel handling accident dose analysis was acceptable. For this amendment request (increase capacity to 3,353 storage locations), the licensee concluded that the analysis was still valid because no parameters of the analysis were affected by the increase in storage capacity. After reviewing the licensee's current submittal and the 1988 safety evaluation, the staff finds the licensee's conclusion to be acceptable. Therefore, the staff finds the proposed SFP expansion to be acceptable.

2.5 Criticality Evaluation

The analysis of the reactivity effects of fuel storage in the Vermont Yankee racks was performed with both the CASMO-4 two-dimensional transport theory code and the MCNP-4A Monte Carlo computer code. CASMO-4 was also used for burnup calculations and to evaluate small reactivity increments associated with manufacturing tolerances and pool temperature changes. MCNP-4A was used to determine reactivity effects of eccentric fuel positioning and fuel assembly drop accidents, fuel misloading outside the racks, and having two BWR racks adjacent to each other. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the Vermont Yankee spent fuel racks as realistically as possible with respect to important parameters such as enrichment, assembly spacing, and absorber thickness. In addition, these two independent methods of analysis (MCNP-4A and CASMO-4) showed very good agreement with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the Vermont Yankee storage racks with a high degree of confidence.

General Design Criterion (GDC) 62 of Appendix A to 10 CFR Part 50 states that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations. This requirement is met by conforming to the NRC acceptance criterion for criticality which states that the effective neutron multiplication factor (k_{eff}) in the spent fuel pool storage racks, if fully flooded by unborated water, shall be no greater than 0.95, including uncertainties at a 95-percent probability, 95-percent confidence level (95/95).

The criticality analyses were performed with several assumptions which tend to maximize the rack reactivity. These include:

(1) Racks contain most reactive fuel authorized to be stored without any control rods or any uncontained burnable poison, and with the fuel at the burnup corresponding to the highest reactivity during its burnup history.

(2) Unborated pool water at the temperature yielding the highest reactivity (4 °C) over the expected range of water temperatures.

(3) Assumption of infinite array (no neutron leakage) of storage cells except for certain accident assessments.

(4) Neutron absorption in minor structural material was neglected (i.e., spacer grids are analytically replaced by water).

(5) The narrowest panel width of the three existing Boral panels was used.

(6) The B-10 density in the Boral panels was assumed to be the minimum value in all panels.

(7) Uniform average U-235 enrichments were used for all fuel rods in a fuel assembly instead of distributed enrichments.

The staff concludes that appropriately conservative assumptions were made.

The design basis fuel assemblies used for the criticality analyses were the standard BWR assemblies with arrays of 7x7, 8x8, 9x9, and 10x10 fuel rods containing UO_2 clad in Zircaloy. For uncertainty calculations, the 10x10 assembly containing 92 fuel rods with 2 water rods was used. An initial uniform U-235 enrichment of 4.6 weight percent (w/o) was assumed. The analysis was performed at the maximum reactivity over burnup, which was found to occur at approximately 11,000 MWD/MTU. At this point, the assembly was analytically transferred into the storage rack at a reference temperature of 4 °C and its k_{eff} in the rack geometry was determined. The same assembly was also analytically transferred into the Vermont Yankee standard cold-core geometry configuration, which is an infinite lattice with 6-inch spacing at a temperature of 20 °C without burnable absorber or control rods and no voids, and its k_{inf} in this cold core configuration was determined. All xenon which was present during the depletion calculations was removed for the rack and cold-core analyses.

For the nominal storage cell design, uncertainties due to boron loading tolerances, boral width tolerances, tolerances in cell lattice spacing and inner dimension, stainless steel thickness tolerances, eccentric positioning, zirconium flow channel bulging, and fuel enrichment and density tolerances were accounted for. These uncertainties were appropriately determined at least at the 95/95 probability/confidence level. In addition, a reactivity uncertainty in the depletion calculation and a 0.01 Δk allowance for possible differences between fuel vendor calculations and those reported here were included. The maximum calculated CASMO-4 reactivity was obtained for the 10x10 fuel and resulted in a k_{eff} of 0.9280 for the new Holtec racks and 0.9469 for the existing Vermont Yankee racks when combined with all known uncertainties. This meets the staff's criterion of k_{eff} no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable.

Based on these results, a BWR fuel assembly appropriate for use in the Vermont Yankee reactor is acceptable for storage in the Vermont Yankee storage racks if it has a k_{inf} in the standard Vermont Yankee core geometry, calculated at the maximum over burnup, of less than or equal to 1.33. Therefore, the requirement incorporated into the Vermont Yankee TS 5.5, "Spent and New Fuel Storage", specifying a k_{inf} of 1.31 remains acceptable for both the existing Vermont Yankee racks and the new Holtec storage racks since it bounds both types.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, such as the accidental insertion of an assembly outside and adjacent to the fuel storage rack, dropping an assembly on top of the rack, or lateral rack movement during a seismic event, which could lead to an increase in reactivity. However, such events were found to have a negligible effect and the resulting reactivity would remain below the 0.95 design basis for both the existing Vermont Yankee racks and the new Holtec storage racks.

The number of fuel assemblies specified in TS 5.5 has been increased from 2870 to 3353 as a result of the requested spent fuel pool expansion. Based on the above evaluation, the NRC staff finds this change to be acceptable.

2.5.1 Conclusion

Based on the review described above, the staff finds the criticality aspects of the proposed modifications to the Vermont Yankee spent fuel pool storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Vermont State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on December 20, 1999 (64 FR 71155). Based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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