SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING THE SPENT FUEL POOL EXPANSION

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 that could be stored in the spent fuel pool from 2000 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 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, and a large number of similar designs have been approved by the NRC.

On May 20, 1988, the NRC staff issued Amendment No. 104, granting the proposed amendment in part: i.e., authorizing 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.

In connection with Amendment No. 104, the staff issued a Safety Evaluation. The staff has incorporated the discussion in that Safety Evaluation in the Safety Evaluation being issued today, which addresses the total request, including the storage of 2870 assemblies and a new enhanced spent fuel pool cooling system.

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. As of April 1986, the licensee had 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; February 25, March 19, March 31, April 9, April 13, May 22, June 11, September 1, and December 11, 1987; and March 2 and June 7, 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 expand the capacity of its spent fuel pool by reracking, 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 beyond the capacity available if the return line were not cut. The licensee stated that the natural circulation developed by the heat generated by the spent fuel would 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 would be minimized, without awaiting completion of review with respect to enhanced cooling, on May 20, 1988 the staff issued Amendment No. 104, which considered the portion of the proposed expansion involving reracking and placement of the new racks in the pool. The staff stated that consideration of storage of more than 2000 assemblies would 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, for which more information was required than was available at the time. Subsequently, on June 7, 1988, the licensee supplemented its application with a document describing in more detail its plans for the new enhanced spent fuel cooling system. The NRC staff's evaluation includes a review of this document.

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 Δk 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 of 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 YAEC 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_∞ value of 1.35. YAEC proposed, in the initial submittal, to use a fuel assembly k_∞ of 1.35 as the design bases for fuel acceptable for storage in the racks (rather than 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 Δk 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_∞ and 3) uncertainty in average assembly enrichment were examined and quantified by YAEC, providing the additional correction factor of 0.04 Δk.

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 Δk to the staff required Technical Specification limit of 0.95. The transfer to a fuel assembly k_∞ design basis criterion has conservatively considered relevant additional uncertainty factors [illegible]d the resulting design basis k_∞ value of 1.31 is acceptable. [illegible]ach of using a k_∞

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 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 deep 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 analyses are performed utilizing the

widely-used industry ANSYS code. Friction between rack supp and pool floor and hydro-dynamic coupling are considered in the analysis. Calculated stresses for the rack components were found 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 pool is 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 21, 1986. 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 < 1μ 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 radiolitic 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 should 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 such as, 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 is, 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 set a dose goal of 23 person-rem for the SFP modification project before committing to add an enhanced fuel pool cooling system. The goal is based on information gained by reviews of the experience gained with similar

projects at other plants. The redundant, seismically designed spent fuel pool cooling system, which would be operational prior to the time Vermont Yankee exceeds the existing 2000 spent fuel assembly storage limits, was proposed by the licensee to resolve all remaining staff concerns related to increasing the storage limit. By telephone conversations on July 7, 1988 the licensee informed the staff that the dose for installation of the enhanced spent fuel pool cooling system has been estimated very conservatively to add less than 10 person-rem to the original dose goal. This results in a dose goal for the entire SFP modification, including the enhanced SFP cooling system, of 33 person-rem. The staff finds this dose goal will not affect the licensee's ability to maintain individual occupational doses within the limits of 10 CFR 20, and as low as is reasonably achievable (ALARA). Normal radiation control procedures, in accordance with the guidelines of Regulatory Guide 8.18, should preclude any significant occupational radiation exposures.

The 33-person 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 of corrosion products, which is 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 first moved 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 foreign material that might become activated 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. Thus, we find that the proposed storage of spent fuel in the modified SFP will not result in any significant dose to workers.

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 were disabled as part of a 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, like other nuclear plants, employs 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 taken 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 on personnel of any accident or abnormal condition 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 33 person-rem for the proposed spent fuel pool expansion are 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 C.F.R. 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 is 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 environmental impact beyond 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 handling 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 Plar 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 for 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/m^3, the effective radiological doses to the control room are estimated to be negligible.

The assumptions used for this analysis are listed 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/m³
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 Sparger Removal

The licensee 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 as originally installed, 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 spargers removed, the water enters and exits 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, 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 demineralizer, 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 administrative 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). 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 an elevated pool water temperature of 200°F.

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

3.11 ENHANCED FUEL POOL COOLING SYSTEM

By letter dated June 7, 1988, VYNPC described a redundant seismically qualified SFPCS, which they will install, test, and make operational prior to the time Vermont Yankee exceeds the current Technical

Specification limit of 2,000 spent fuel assemblies stored in the Vermont Yankee Spent Fuel Pool.

The enhanced system proposed by Vermont Yankee is the Fuel Pool Cooling And Demineralizer System, which consists of two subsystems: (1) Normal Fuel Pool Cooling System (NFPCS), which is the existing cooling system, and (2) a new additional cooling system called the Emergency Standby Subsystem (ESS). Both subsystems have two trains of pumps and heat exchangers.

The NFPCS provides the normal spent fuel pool cleaning and cleanup. It is non-seismic Category I, with one 450 gpm centrifugal pump and one 2.23 MBtu/hr tube-and-shell heat exchanger in each train. The NFPCS cools the spent fuel pool by transferring the spent fuel decay heat through heat exchanger(s) to the Reactor Building Closed Cooling Water System. The NFPCS cleans the pool water by circulating the water through the filters and demineralizers to provide pool water clarity and purity. One train of the NFPCS is normally lined up and operating during plant operation.

The ESS will be used to cool the spent fuel pool when a high spent fuel decay heat load is placed in the pool, or if a seismic event occurs. The ESS is seismic category I, with one 700 gpm pump and one 11.0 MBtu/hr tube-and-shell heat exchanger in each train. The ESS equipment receives electrical power from safety-related IE power sources. Each train is sized to maintain the pool water temperature

below 150°F with assumed normal refueling heat loads. It cools the pool by transferring the spent fuel decay heat through the heat exchangers to the Service Water System maintaining pool temperature below 150°F. The ESS is normally not in use.

3.11.1 Heat Removal Capability

Based on the guidelines of SRP Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup Systems" two spent fuel pool heat generation conditions were evaluated. The first condi[illegible] was the normal maximum heat load, which is the heat gene[illegible] spent fuel when the pool is filled (2,870 assemblies) [illegible] ssive regular refuelings (1/3 of the core for each refueling interval) assuming a failure of one train of pool cooling. The second condition was the abnormal maximum heat load case, which is similar to the normal maximum heat load case except that the last 368 spaces are filled by a full core off-load and no failures are assumed.

The decay heat loads from normal refuelings were calculated by the licensee with the assumption of discharging the spent fuel to the storage pool at six days and ten days following shutdown from normal operation. The abnormal decay heat load was calculated assuming that the full core discharge to the storage pool occurs ten days following shutdown from normal operation for refueling.

Data in Table A.2 from the licensee's June 7, 1988 submittal indicate that the decay heat generation rate for normal refueling discharge varies from 7.59 MBtu/hr to 10.33 MBtu/hr, and that the rate for the abnormal refueling discharge varies from 16.84 MBtu/hr to 18.26 MBtu/hr (six and ten days respectively). These heat generation rates are comparable to the staff's independent calculated values.

Based upon a comparison of these calculated heat loads and the heat removal capability of the cooling systems, the Fuel Pool Cooling and Demineralizer System has sufficient capacity to remove the calculated maximum normal and abnormal heat generated by the spent fuel.

The Normal Fuel Pool Cooling System (NFPCS) has a heat removal capacity of 2.23 MBtu/hr per train with a pool water temperature of 125°F. Should the NFPCS be unable to maintain fuel pool temperature or if it should lose flow, the Emergency Standby System (ESS) will be used. The ESS has a heat removal capacity of 11.0 MBtu/hr per train. In the normal maximum heat load condition the maximum calculated heat load is 10.33 MBtu/hr as compared with an EES heat removal capability of 11 MBtu/hr per train. In the abnormal maximum heat load condition, the total heat removal capacity is 22.0 MBtu/hr as compared with a maximum calculated heat load of 18.26 MBtu/hr. Therefore, the ESS has sufficient heat removal capability following the methodology as discussed in SRP 9.1.3.

3.11.2 Water Level, Makeup Water and Corrosion

Leakage of potentially radioactive water to the environment from the ESS is prevented by providing a higher pressure in the service water system than the ESS pressure. Indication of this differential pressure is provided in the Control Room. Leakage from the NFPCS to the service water system is prevented by using an intermediate closed loop cooling system, Reactor Building Closed Cooling Water (RBCCW), which transfers the decay heat to the Service Water System. This closed loop system ensures that fuel pool water leakage, if any, will be contained within the RBCCW System and will not be released into the Service Water System.

The Fuel Pool Cooling and Demineralizer System has instruments to monitor the water level and maintain a water level above the fuel sufficient to provide shielding for normal building occupancy. Makeup water to the SFP is normally provided from the Condensate Transfer System or the Demineralized Water System. Makeup to the SFP to account for leakage and evaporation can also be provided by the seismic Category I service water system.

The fuel pool system pumps and heat exchangers in contact with the pool water as well as associated piping and valves are corrosion-resistant material. The filter-demineralizer maintains total heavy element content in the pool at 0.1 ppm or less. Particulate material is removed by the pressure precoat filter-demineralizer unit. A post-strainer is provided in the effluent stream of the filter-demineralizer to limit the migration of the filter material. Two small skimmer pumps are provided to remove surface debris by pumping water from the top of the pool through cartridge filters then back to the pool through the service boxes.

3.11.3 Isolation Capability

Isolation of the non-seismic NFPCS from the ESS is achieved by two check valves V-19-18 and V-19-G and two isolation valves of the NFPCS, V-19-H and -I which are nonthrottling MOVs, each powered by a different safety-related electrical power supply. Thus, General Design Criterion 2, requiring isolation of seismically qualified systems from non-seismic systems is satisfied.

Each of the two heat exchangers in the ESS has a service water outlet MOV. These two MOVs, V-19-J and -K, are throtting-type valves providing service water flow control, and thereby

controlling both the pool temperature and the differential pressure between service water and the fuel pool water.

The ESS is designed to provide pool cooling under all licensed plant conditions. This subsystem is designed as seismic Category I using the seismic Category I Service Water System to remove spent fuel decay heat to the ultimate heat sink (Connecticut River). Essential electrical components are also environmentally qualified to ensure operability under design basis accident conditions. Therefore, pool boiling will not occur, and the Reactor Building environment will not be subject to the consequences of a boiling spent fuel pool.

Instrumentation and controls are provided to detect, control and record pump operation, pool temperature, and system flow. A pool leak detention system is provided to monitor leakage through the pool liner.

3.11.4 Inspection and Testing

The NFPCS normally has one train in operation. Redundant units are operated periodically to handle abnormal heat loads or for maintenance. The redundant units of the ESS are periodically operated to ensure that this subsystem can be isolated and provide cooling by remote manual initiation. Also, routine visual inspections for both subsystems' components,

instrumentation and alarms will be performed to verify system operability.

3.11.5 Summary: Enhanced Fuel Pool Cooling System

The staff has reviewed and evaluated the enhanced fuel pool cooling system as described in Vermont Yankee's submittal dated June 7, 1988. The staff finds that:

- The enhanced spent fuel pool cooling system includes an Emergency Standby Subsystem (ESS) which has seismic Category I redundant cooling trains with a seismic Category I makeup water source.
- The heat removal capacity of one train of the ESS is adequate for normal maximum heat load removal. The heat removal capacity of both trains of the ESS is adequate for the maximum abnormal heat load removal.
- The enhanced system has appropriate instruments to monitor the pool water level, and maintains the necessary water level above the spent fuel bundles.
- The filter-demineralizer of the enhanced system provides the pool water clarity and purity.
- The enhanced system is able to maintain uniform pool water temperature and maintain the temperature at or below the Technical Specification limit of 150°F.

4.0 SUMMARY

Based on these findings the staff concludes that the Vermont Yankee high density racks and fuel pool, including the enhanced system for spent fuel pool cooling is acceptable for the proposed expansion to 2870 assemblies, provided that the number of spent fuel assemblies does not exceed 2000 until the enhanced system as described in the licensee's June 7, 1988 submittal has been installed and tested to demonstrate operability.

5.0 ENVIRONMENTAL CONSIDERATIONS

A separate Environmental Assessment has been prepared pursuant to 10 C.F.R. Part 51. The Notice of Issuance of Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on August 1, 1988 (53FR28925).

6.0 CONCLUSIONS

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

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

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

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