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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 214 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 214 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

In a letter of November 5, 1997, supplemented by letters of January 28, 1998, and May 12, 1998, Virginia Electric and Power Company (VEPCO) requested changes to the Surry Power Station Units 1 and 2 Technical Specifications (TS) to reflect an increase in the U-235 enrichment of fuel stored in the fresh fuel storage racks or the spent fuel storage racks from 4.1 weight percent (w/o) U-235 to 4.3 w/o U-235.

The January 28 and May 12, 1998 submittals provided clarifying information that did not affect the initial no significant hazards determination.

2.0 EVALUATION

2.1 Criticality

The analysis of the reactivity effects of fuel storage in the Surry fresh and spent fuel racks was performed with the three-dimensional Monte Carlo code, KENO-Va, with neutron cross sections generated with the NITAWL and BONAMI codes. Since the KENO-Va code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with the two-dimensional transport theory code, PHOENIX-P. The analytical methods and models used in the reactivity analysis have been benchmarked against experimental data for fuel assemblies similar to those for which the Surry racks are designed and have been found to adequately reproduce the critical values. This experimental data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include close proximity storage and strong neutron absorbers. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the Surry storage racks with a high degree of confidence.

The NRC acceptance criterion for preventing criticality outside the reactor is that, including uncertainties, there is a 95% probability at a 95% confidence level (95/95 probability/confidence) that the effective neutron multiplication factor (k_{eff}) of the fuel assembly array, when moderated by unborated water, will be no greater than 0.95. This k_{eff} limit applies to both the fresh and spent fuel racks, except for the fresh fuel rack under low water density (optimum moderation) conditions, where the k_{eff} limit is 0.98. The analyses for storage of 4.3 w/o U-235 fuel in the Surry fresh and spent fuel storage racks assumed fuel assembly parameters based on the Westinghouse 15x15 fuel assembly design used at Surry.

For the fresh fuel racks, the previous criticality analyses for the North Anna plant evaluated the effects of varying moderator density for the storage of fresh Westinghouse 17x17 fuel assemblies with nominal enrichments up to 4.3 w/o U-235. The North Anna new fuel storage racks have the same design as the Surry racks. However, since Surry uses 15x15 fuel assemblies, the North Anna values were increased by 0.005 Δk to account for the slightly higher reactivity (≈0.5%) of the Surry assemblies relative to the North Anna 17x17 assemblies. Additional uncertainties were applied to account for any fuel or design tolerance uncertainties.

The worst case 95/95 k_{eff} as a function of moderator density reaches a peak value of 0.930 at a water density of 0.07 g/cc. This is the low-density optimum moderation case. Since k_{eff} is less than 0.98, the acceptance criterion for preventing criticality under optimum moderation conditions is met. For the fully flooded accident scenario, the 95/95 k_{eff} for 4.3 w/o enriched fuel is slightly less than this and, therefore, meets the acceptance criterion of 0.95.

The Surry spent fuel storage pool is categorized into two regions, referred to as Region 1 and Region 2. Each storage cell is constructed of type 304 stainless steel having exterior dimensions of 9.12 inches and a wall thickness of 0.090 inches. These cells have a 14-inch center-to-center spacing and contain no neutron absorber panels. Region 1 contains 324 storage locations and currently provides storage for irradiated fuel with an initial enrichment up to 4.1 w/o U-235, with restrictions based on assembly burnup as given in TS Figure 5.4-1. Region 2 contains 720 storage locations and can presently contain fuel up to 4.1 w/o U-235 without any restrictions on assembly burnup.

The Surry spent fuel racks have been reanalyzed to allow an increase in the maximum enrichment to 4.3 w/o U-235, based on a nominal fresh reference enrichment of 4.25 w/o U-235 and an enrichment tolerance of ±0.05 w/o. For the nominal storage cell design, the moderator was assumed to be pure water at a temperature of 170°F, which is the most reactive condition over the normal range of pool water temperatures. Uncertainties due to tolerances in fuel enrichment and density, storage cell inner dimension, storage cell center-to-center pitch, stainless steel thickness, assembly position, as well as a benchmarking bias uncertainty and a calculational uncertainty were accounted for. These uncertainties were appropriately determined at least at the 95/95 probability/confidence level. In addition, a methodology bias (determined from benchmark calculations) as well as an allowance for uncertainties in depletion calculations for those cases where burnup credit is used, were included. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

The licensee's analysis using the acceptable methods discussed above has shown that fuel assemblies containing a 15x15 rod array with maximum enrichments up to 4.3 w/o U-235 result in a spent fuel pool k_{eff} of 0.944. Since this meets the NRC acceptance criterion of no greater than 0.95, fuel with a maximum U-235 enrichment of 4.3 w/o can be stored in any location in Region 2 without regard to minimum burnup.

Although the entire fuel pool contains the same rack design, a fuel storage cask handling accident scenario must be considered for the first three rows of fuel storage racks adjacent to the fuel building trolley load block, designated as Region 1. To evaluate the impact of a storage cask handling accident on criticality, the fuel assemblies in these rows were assumed to be crushed and the deformed fuel and the associated storage racks were assumed to be at the optimum pitch. The assembly to assembly spacing was reduced from 14 inches to approximately 6.9 inches. This spacing assumes contact between the fuel storage rack cells and a uniform reduction of the fuel assembly fuel pin pitch. KENO-Va calculations determined that the maximum fresh fuel enrichment under these optimum pitch assumptions that meets the 95/95 0.95 k_{eff} limit was 1.9 w/o U-235, with credit for 2250 ppm of soluble boron in the pool water. It is unnecessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident (double contingency principle). Therefore, the presence of soluble boron in the pool water can be assumed as a realistic initial condition during a cask drop since assuming its absence would be a second unlikely event. Also, since TS 5.4.C requires at least 2300 ppm of boron in the water whenever there is spent fuel stored in the pool, the assumption of 2250 ppm of boron for accidents is acceptable.

The Region 1 racks, fully loaded with fuel enriched to 1.9 w/o U-235 and damaged as a result of a cask drop accident, resulted in a pool k_{eff} of 0.93204, including all applicable uncertainties and tolerances, with 2250 ppm of soluble boron in the pool water. To allow the loading of fuel with maximum enrichments up to 4.3 w/o U-235 in Region 1 of the Surry spent fuel pool, credit was taken for the fuel burnup using the standard reactivity equivalencing methodology approved by the NRC. In this case, fuel assemblies stored in Region 1 must satisfy the minimum burnup versus initial enrichment requirements specified in the proposed TS Figure 5.4-1, which shows that fuel initially enriched to 4.3 w/o U-235 must achieve a burnup of approximately 33,000 MWD/MTU. TS 5.4.B requires that administrative controls with written procedures be employed in the selection and placement of these assemblies.

Most abnormal storage conditions will not result in an increase in the k_{eff} of either the fresh fuel or the spent fuel storage in the racks. However, it is possible to postulate events, such as flooding the dry fresh fuel storage racks, the inadvertent misloading of an assembly in the spent fuel storage racks with a burnup and enrichment combination outside of the acceptable areas in TS Figure 5.4-1, or a pool water temperature change, which could lead to an increase in reactivity. Flooding of the fresh fuel racks under fully flooded or optimum moderation conditions was shown above to meet the limiting k_{eff} of 0.95 and 0.98, respectively. For the spent fuel pool accidents, the double contingency principle allows credit to be taken for the presence of at least 2300 ppm of soluble boron in the pool water equivalent to that used in the reactor cavity and refueling canal during refueling operations, which is assured by TS 5.4.C. Except for the cask drop accident previously discussed, the reduction in k_{eff} caused by the boron more than offsets the reactivity addition caused by credible accidents.

The following Technical Specification changes have been proposed as a result of the requested enrichment increase. Based on the above evaluation, the staff finds these changes acceptable.

1) TS 5.3.A.2 has been changed to increase the maximum enrichment of reload fuel from 4.1 w/o to 4.3 w/o U-235.

2) TS 5.4.B has been changed to increase the maximum enrichment of the fuel which may be stored in the spent fuel racks from 4.1 to 4.3 w/o U-235.

3) TS Figure 5.4-1 has been changed to allow fuel enriched to 4.3 w/o U-235 to be stored in Region 1 with burnup restrictions and 2250 ppm of boron in the pool water.

Based on the review described above, the staff finds that the criticality aspects of the proposed increase in the fuel enrichment limit of the Surry fresh and 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.

Although the Surry TS have been modified to specify the above-mentioned fuel as acceptable for storage in the spent fuel racks, evaluations of reload core designs (using any enrichment) will be performed on a cycle-by-cycle basis as part of the reload safety evaluation process. Each reload design is evaluated to confirm that the cycle core design adheres to the limits that exist in the accident analyses and TS to ensure that reactor operation is acceptable.

2.2 Decay Heat Removal

In letters dated November 5, 1997, January 28, 1998, and May 12, 1998, the licensee stated they had reviewed the impact of the proposed enrichment increase on fuel management and determined the impact on the spent fuel decay heat load. The licensee stated they had confirmed by calculation that the decay heat load analysis described in the Surry UFSAR remains bounding for the anticipated fuel management with the slightly higher enrichment. The staff agrees that the small increase in enrichment (0.2 weight percent) will have a very minor impact on decay heat load and finds the decay heat removal capability acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comment.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on March 17, 1998 (63 FR 13079). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of the 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: L. Kopp

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