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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO FUEL ENRICHMENT INCREASE
SYSTEM ENERGY RESOURCES, INC.
GRAND GULF NUCLEAR STATION UNIT 1
DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated October 27, 1988, System Energy Resources, Inc., (SERI) submitted a criticality analysis for storage of Grand Gulf Nuclear Station, Unit 1 (GGNS-1), Cycle 4 reload fuel in the spent fuel racks (Ref. 1). The current GGNS-1 spent fuel storage racks are analyzed for storage of 8x8 fuel assemblies containing up to 3.5 weight percent (w/o) uniformly enriched U-235. The Cycle 4 8x8 fuel design contains an average enrichment of 3.61 w/o U-235 based on the specific enrichment distribution. In addition, four lead test assemblies (LTAs) of 9x9-5 fuel will be loaded in the core for Cycle 4 operation. Therefore, the licensee also submitted criticality analysis for the 9x9-5 LTAs (Ref. 2).

2.0 EVALUATION

The GGNS-1 spent fuel storage racks consist of an egg-crate structure with fixed neutron absorber material (Boraflex) of 0.02041 grams B-10 per square centimeter positioned between the stainless steel walls of each cell. Each storage cell has an inside dimension of 6.0625 inches and provides a nominal center-to-center cell spacing of 6.259 inches.

The KENO-IV Monte Carlo computer code was used to calculate the reactivity of the GGNS storage racks when fully loaded with the Cycle 4 8x8 fuel. Neutron cross section data from the 27-group SCALE library was generated for input to KENO-IV using the AMPX code. The KENO-Va code was used to calculate the reactivity of the storage racks when fully loaded with Cycle 4 9x9-5 LTAs. Cross sections were generated from the 16-group SCALE library using the BONAMI code. Small incremental reactivity effects were calculated with the two-dimensional, multi-group, transport theory codes CASMO-2E and CASMO-3. These models have been benchmarked against experimental data and have been found to adequately reproduce the critical values.

The spent fuel pool criticality calculations were conservatively based on replacing gadolinia with uranium, no credit for radial neutron leakage, a moderator temperature corresponding to the maximum reactivity within the design range, and neglecting the neutron absorption of minor structural members. Reactivity effects of manufacturing tolerances were determined at the 95% probability 95% confidence (95/95) level, thereby meeting the NRC requirements. The maximum reactivity of the spent fuel storage racks was calculated to be 0.936, including the calculational bias and all known

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uncertainties due to manufacturing tolerances at the 95/95 probability/confidence level, when fresh 8x8 fuel of the Cycle 4 enrichment distribution is stored therein. The maximum reactivity of the racks fully loaded with the 9x9-5 LTAs was calculated to be 0.9197, including the appropriate bias and 95/95 uncertainties. These results meet the staff's acceptance criterion of k_{eff} no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level.

The criticality calculations were based on the actual Cycle 4 enrichment distribution which results in an average fuel assembly enrichment of 3.61 w/o U-235 for the 8x8 fuel and 3.47 w/o U-235 for the 9x9-5 LTAs. However, since calculations based on a uniform 3.61 w/o U-235 enrichment (i.e., each fuel rod enriched to 3.61 w/o U-235) resulted in a k_{eff} approximately 0.011 delta-k higher than the corresponding calculation with distributed enrichments. Any future changes in enrichment distribution should be reviewed by the licensee for assurance that the 0.95 acceptance criterion for spent fuel storage is met.

The effects of abnormal and accident conditions were evaluated and found to meet the staff's acceptance criterion of k_{eff} no greater than 0.95. These conditions included positioning an assembly outside the storage rack, positioning an assembly in an eccentric location within the storage cells, dropping a fuel assembly, laterally moving a rack module, and removing a zirconium flow channel.

The staff requested additional information from the licensee concerning possible Boraflex degradation and the measuring techniques in place at GGNS-1 for detecting this (Ref. 4). As a result of augmented surveillance testing, the licensee has observed gaps in some of the Boraflex panels in the spent fuel racks. This, of course, reduces the neutron absorption effectiveness of the Boraflex and may cause a decrease in the margin of subcriticality of the fuel pool. Calculations by the licensee assuming a 3-inch wide gap at the axial midplane of every Boraflex panel resulted in a k_{eff} of 0.947 (Ref. 3), which meets the k_{eff} acceptance criterion of 0.95. Blackness test measurements have confirmed the presence of gaps in about 40% of the 212 irradiated panels tested with an average gap size of 0.8 inch and a maximum gap size of 1.4 inches. Therefore, the calculation assuming a 3-inch gap in every panel is conservatively bounding at present and indicates the staff's acceptance criterion is not violated. Further calculations have shown that even if the presence of the maximum observed gap size had been explicitly accounted for in the Cycle 4 spent fuel pool criticality analyses, the reported reactivity for both the 8x8 and the LTA fuel would essentially be unchanged. The licensee is performing a criticality analysis based on the number and size of gaps projected to exist at the time of the next planned blackness tests. This analysis is scheduled for completion in February 1989. The licensee will monitor (by blackness testing) approximately 50 storage cells during each cycle. These cells will be representative of those which have received the highest amount of gamma radiation and so should maximize the formation of gaps. The analysis will provide confirmation of the acceptability of the proposed Boraflex surveillance time interval. In the interim, the staff concurs that the previous analysis, which assumed a 3-inch wide gap in every panel, remains conservatively bounding and that the surveillance program in place at GGNS-1 will provide an early indication of any significant Boraflex degradation and a confirmation of any assumptions used in the criticality analyses. If, as a result of ongoing Boraflex studies and surveillance tests described in Reference 4, existing gap sizes increase or additional gaps develop which may affect the required subcriticality margin, the licensee will be required to perform additional analyses or provide corrective actions to justify continued storage in the GGNS-1 spent fuel pool.

3.0 CONCLUSIONS

Based on the review described above, the staff finds the criticality aspects of the storage of Cycle 4 8x8 reload fuel in the GGNS-1 spent fuel storage racks acceptable. Storage of the 9x9-5 LTAs is also acceptable. However, because the criticality calculations were based on the actual Cycle 4 enrichment distribution, any future changes in enrichment distribution should be reviewed by the licensee to assure that the 0.95 acceptance criterion for spent fuel storage is still met.

The results of the Boraflex analysis scheduled for completion on February 15, 1989, and the detailed Boraflex surveillance program should be submitted to the NRC by February 28, 1989. In addition, if additional Boraflex degradation is observed which could decrease the margin of subcriticality of the fuel pool below the licensing basis calculation, the licensee will be required by 10 CFR 50.59 to perform additional analyses to verify that the required 5% subcriticality margin can be maintained or provide corrective actions to justify continued storage in the GGNS-1 spent fuel pool.

4.0 REFERENCES

1. Letter from W. T. Cottle (SERI) to NRC submitting "Criticality Safety Analysis of Cycle 4 Fuel in the Grand Gulf Nuclear Station Spent Fuel Racks," AECM-88/0206, October 27, 1988.

2. Letter from T. H. Cloninger (SERI) to NRC submitting "Criticality Safety Analysis for the Grand Gulf Spent Fuel Storage Racks with Cycle 4 9x9-5 LTA (3.47% Average Enrichment)," AECM-88/0228, November 15, 1988.

3. Letter from J. G. Cesare (SERI) to NRC submitting "Response to RAI Regarding Boraflex Gap Analysis," AECM-88/0237, November 30, 1988.

4. Letter from J. G. Cesare (SERI) to NRC submitting "Response to NRC Question Regarding GGNS Boraflex Gap Surveillance Program," AECM-88/0233, November 21, 1988.