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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. NPF-8

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-348 AND 50-364

INTRODUCTION

By letter dated March 30, 1984, supplemented May 29, 1984, Alabama Power Company (APCo) requested changes to the Joseph M. Farley Nuclear Plant Units 1 and 2 Technical Specifications which would increase the fuel enrichment limit form 3.5 to 4.3 weight percent U-235 for new fuel storage and use of new fuel in the core. These changes are requested in order to accommodate the fuel necessary for implementation of 18-month fuel cycles. APCo states that the evaluation of fuel with greater than 3.5 weight percent in the reactor will be made on a cycle specific basis as part of the reload safety evaluation process. A report, "Summary Report Nuclear Criticality Re-Analysis for 4.3 w/o Fuel in New Fuel Storage Rack of Joseph M. Farley Nuclear Plant of Alabama Power Company" by Utility Associates International was submitted in support of the change to the new fuel storage racks. Our discussion and evaluation follows:

DISCUSSION AND EVALUATION

1. Analysis Method

The criticality aspects of the storage of new fuel rods was analyzed using the multigroup, two-dimensional, transport theory code CASMO-2E and the Monte-Carlo transport model KENO-IV/AMPX. The KENO-IV/AMPX code system has been benchmarked against several critical experiments. The results of two particular experiments containing no boron were compared with the KENO-IV/AMPX predictions. The KENO-IV/AMPX results were -0.001 and 0.002 Δk below critical

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values for the two experiments. The CASMO code has been benchmarked against the KENO-IV/AMPX code for several configurations. The results from 16 configurations showed CASMO results over-predict the KENO-IV/AMPX results by 0.005 Δk overall.

2. New Fuel Storage Rack Analysis

Although new fuel is normally stored in a dry configuration, the NRC acceptance criteria for new fuel storage is that there is a 95 percent probability at a 95 percent confidence level (including uncertanties) that k_{eff} of the fuel assembly array will be; (1) no greater than 0.95 when fully loaded and flooded with unborated water and (2) no greater than 0.98 under conditions of low density (optimum moderation) if higher reactivities can be attained at achievable moderation conditions other than full density unborated water.

The CASMO 2E model used 0.1% density water to predict the upper limit of k_∞ for the dry case at 68°F. This was done for an infinite lattice configuration with no boron present. The result was $k_\infty = 0.8883$. Since this case did not include leakage, the true k_{eff} for the dry case would be considerably less. CASMO 2E was used to determine the k_∞ vs water density curve over a range of 0.001 to 1 gm/cm^3.

Reference cases using nominal rack geometry and 2% and 5% water density were done with KENO-IV. These cases demonstrate that the finite rack configuration is substantially subcritical for all water densities even though no credit was taken for axial leakage. The results were:

	k_{eff}	95% confidence level
k_{eff} 2% density	0.7094 ± .0106	0.6882 to 0.7306
k_{eff} 5% density	0.7480 ± .0113	0.7254 to 0.7706

3. Uncertainties and Tolerances

Uncertainties and tolerances consist of three things:

1. 95% confidence level
2. the bias between KENO-IV and measurements
3. the bias due to positional and dimensional tolerances.

Using the worst of these biases and uncertainties the k_{eff} for the 5% water density case was 0.796 which is much less than the 0.98 level for optimum moderation.

4. Accident Considerations

The two accident conditions considered were flooding of the new fuel pit and a dropped assembly between the periphery of the new fuel rack and the fuel pit wall. For the flooded case the CASMO-2E result was k_∞ = 0.8160 for the infinite lattice configuration. Since this case did not include leakage, the k_{eff} of the fully flooded case would be considerably lower. The dropped assembly case was shown to have a k_{eff} less than 0.883. Thus, these events were shown to have k_{eff} values much less that the acceptance criterion noted in paragraph 3. above.

5. Reactor Core Fuel Assemblies

The licensee requested a change to the maximum reactor core fuel enrichment from 3.5 weight percent U-235 to 4.3 weight percent in the Technical Specification 5.3. Fuel enrichment is not a direct input to the reactor safety analysis. Fuel enrichment is used in conjunction with a number of parameters and considerations in determining safe operation of the reactor. The fuel enrichment, number of fuel assemblies, exposure (burnup) of existing fuel, burnable poisons, and fuel management schemes are used to derive measurable reactor core parameters important to safe operation. These dynamic parameters such as shutdown margin, reactivity coefficients, and power peaking factors are included in the Technical Specifications. The specification of the fuel enrichment in the core design section alone does not uniquely determine nor limit the values of the reactor core parameters which are important for safe operation.

SAFETY SUMMARY

Based on our review of the licensee proposals and the discussion and evaluation contained herein, we conclude that fuel assembies with a maximum enrichment of 4.3 weight percent U-235 can be stored safely in the new fuel racks at Farley Nuclear Plant Units 1 and 2. Our conclusion is based on the following:

1. Criticality calculations were performed with acceptable models and methods.
2. Unertainties have been accounted for as described above.
3. Postulated accidents have been considered.
4. The multiplication factor, including uncertainties, meets our acceptance criteria for this quantity.

Based on our review, we find the proposed change of the enrichment restriction in the Technical Specification 5.3 is also acceptable.

Environmental Consideration

These amendments involve a change in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR 20. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

Conclusion

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

Dated: November 8, 1984

Principal Contributors: