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

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 57 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 49 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

Alabama Power Company (APCo) by letter dated February 3, 1984, supplemented September 14, and November 26, 1984, requested amendments to the Farley Units 1 and 2 Technical Specifications (TSs). The proposed amendments consist of revisions to the Technical Specifications proposed pursuant to NRC Generic Letter 84-43 to assure compliance in reporting events to be in compliance with 10 CFR 50.72 and 10 CFR 50.73. Our evaluation follows.

Evaluation and Discussion

Generic Letter (GL) 83-43, dated December 19, 1983, informed utilities of changes to Title 10 of the Code of Federal Regulations (10 CFR) with respect to notification requirements for operating nuclear power reactors and requested utilities to propose changes to Technical Specifications to incorporate CFR changes. The CFR changes involve a revision to Section 50.72 of 10 CFR for the immediate notification requirements. A new Section 50.73 in 10 CFR revised the Licensee Event Report System. Both of these changes became effective January 1, 1984. GL 83-43 provided a model Technical Specification showing the revisions which should be made in the "Administrative Control" and "Definitions" sections of Technical Specifications to incorporate these regulation changes.

By letter dated February 3, 1984, APCo responded to GL 83-43 with proposed changes to "Administrative Controls" and "Definitions" only. Our review indicated that supplemented information was necessary from the licensee. By letter dated July 25, 1984, we requested additional information which APCo provided by letters dated September 14 and November 26, 1984. These supplementary letters provided the additional information and other conforming changes required to reflect the revised reporting requirements throughout the entire Technical Specifications as stated in GL 83-43. The additions are considered non-substantive to the licensee's February 3, 1984 submittal which the Commission had prenoticed. However, these changes were necessary to assure full compliance with Commission guidance promulgated by GL 83-43 and 10 CFR 50.72 and 10 CFR 50.73 in the other parts of the Technical Specifications where reporting requirements were referenced.

Safety Summary

Based on our review of the licensee responses to GL 83-43 as noted above, we find the proposed changes to the Farley Units 1 and 2 Technical Specifications to be acceptable. We have determined that no significant hazards consideration is involved since the changes are administrative in nature.

Environmental Consideration

These amendments involve only changes in administrative procedure and requirements. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(10). 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: February 19, 1985

Principal Contributors:

F. R. McCoy
E. A. Reeves

ATTACHMENT TO LICENSE AMENDMENT NO. 52

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

DOCKET NO. 50-348

Revise Appendices A and B as follows:

Remove Pages	Insert Pages
Appendix A	Appendix A
I	I
XIX	XIX
XX	XX
1-6	1-6
3/4 0-1	3/4 0-1
3/4 1-4	3/4 1-4
3/4 2-6	3/4 2-6
3/4 3-59	3/4 3-59
3/4 3-61	3/4 3-61
3/4 3-66	3/4 3-66
3/4 4-24	3/4 4-24
3/4 6-9	3/4 6-9
3/4 7-81	3/4 7-81
3/4 7-82	3/4 7-82
3/4 7-85	3/4 7-85
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3/4 7-90	3/4 7-90
3/4 7-92	3/4 7-92
3/4 7-94	3/4 7-94
3/4 8-6	3/4 8-6
3/4 11-4	3/4 11-4
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3/4 11-17	3/4 11-17
3/4 11-19	3/4 11-19
3/4 12-1	3/4 12-1
3/4 12-9	3/4 12-9
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B3/4 2-5	B3/4 2-5
B3/4 4-3	B3/4 4-3
6-8	6-8
6-10	6-10
6-14	6-14
6-15a	6-15a
6-19	6-19
6-20	none
6-21	none
6-22	6-20
6-23	6-21
6-24	6-22
6-25	6-23
6-26	6-24
6-27	6-25
Appendix B	Appendix B
4-1	4-1

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10CFR Part 50.

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

SOURCE CHECK

1.30 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.31 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.