October 15, 1984

Docket Nos. 50-348
and 50-364

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Mr. R. P. McDonald
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

Dear Mr. McDonald:

The Commission has issued the enclosed Amendment No. 50 to Facility Operating License No. NPF-2 and Amendment No. 41 to NPF-8 for the Joseph M. Farley Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated April 10, 1984.

The amendments modify the Technical Specifications to define the reactor coolant system pressure isolation valves (PIV's) allowable leakage criteria per Commission guidance contained in License Amendment No. 25 for Farley Unit 2, dated September 8, 1982. The change also standardizes the Technical Specifications for Farley Unit Nos. 1 and 2 in accordance with additional Commission guidance relating to the number and identification of PIV's to be tested.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

/s/EReeves

Edward A. Reeves, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 50 to NPF-2
2. Amendment No. 41 to NPF-8
3. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL	ORB#1:DL	C-ORB#1:DL	OELD	AD:OR:DL
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Mr. R. P. McDonald
Alabama Power Company

Joseph M. Farley Nuclear Plant
Units 1 and 2

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

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
License No. NPF-2

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Alabama Power Company (the licensee) dated April 10, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-2 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 50 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 15, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 50

AMENDMENT NO. 50 FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Revised Appendix A as follows:

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.4 REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 31 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 $\pm$ 20 psig.

f. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve shall be as specified in Table 3.4-1 at a pressure of 2235 $\pm$ 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit specified in Table 3.4-1, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by;

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment air cooler condensate level system or containment atmosphere gaseous radioactivity monitor at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals at least once per 31 days when the Reactor Coolant System pressure is 2235 $\pm$ 20 psig with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.7.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5 except that in lieu of any leakage testing required by Specification 4.0.5, each valve should be demonstrated OPERABLE by verifying leakage to be within the allowable leakage criteria of 0.5 gpm per inch of nominal valve size with an upper limit of the maximum allowable leakage in Table 3.4-1; and the measured leak rate for any given test cannot reduce the difference between the results of the previous test and the maximum allowable leakage specified in Table 3.4-1 by more than 50%:#

a. Every refueling outage during startup.

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve affecting the seating capability of the valve.

c. Following valve actuation due to automatic or manual action or flow through the valve for valves identified in Table 3.4-1 by an asterisk.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

To satisfy ALARA requirements, leakage may be measured indirectly (as from performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	DESCRIPTION	MAXIMUM ALLOWABLE LEAKAGE**
Q1E11V001A	12" GATE	5.000 GPM
Q1E11V001B	12" GATE	5.000 GPM
Q1E11V016A	12" GATE	5.000 GPM
Q1E11V016B	12" GATE	5.000 GPM
Q1E11V021A	6" CHECK	3.000 GPM
Q1E11V021B	6" CHECK	3.000 GPM
Q1E11V021C	6" CHECK	3.000 GPM
* Q1E21V032A	12" CHECK	5.000 GPM
* Q1E21V032B	12" CHECK	5.000 GPM
* Q1E21V032C	12" CHECK	5.000 GPM
* Q1E21V037A	12" CHECK	5.000 GPM
* Q1E21V037B	12" CHECK	5.000 GPM
* Q1E21V037C	12" CHECK	5.000 GPM
Q1E11V042A	10" CHECK	5.000 GPM
Q1E11V042B	10" CHECK	5.000 GPM
* Q1E21V076A	6" CHECK	3.000 GPM
* Q1E21V076B	6" CHECK	3.000 GPM
* Q1E21V077A	6" CHECK	3.000 GPM
* Q1E21V077B	6" CHECK	3.000 GPM
Q1E21V077C	6" CHECK	3.000 GPM

* Indicates the requirements of Section 4.4.7.2.2 Item (c) are applicable.

** The measured leak rate for any given test cannot reduce the difference between the results of the previous test and the maximum allowable leakage specified in Table 3.4-1 by more than 50%.

3/4.4.7 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.7.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.7.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 31 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation valves is IDENTIFIED LEAKAGE and will be considered a portion of the allowed limit.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

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

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
License No. NPF-8

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Alabama Power Company (the licensee) dated April 10, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-8 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 41 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 15, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 41

AMENDMENT NO. 41 FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Revised Appendix A as follows:

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OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 31 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 $\pm$ 20 psig.

f. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation Valve shall be as specified in Table 3.4-1 at a pressure of 2235 $\pm$ 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit specified in Table 3.4-1, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by;

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment air cooler condensate level system or containment atmosphere gaseous radioactivity monitor at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals at least once per 31 days when the Reactor Coolant System pressure is 2235 ± 20 psig with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.7.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5 except that in lieu of any leakage testing required by Specification 4.0.5, each valve should be demonstrated OPERABLE by verifying leakage to be within the allowable leakage criteria of 0.5 gpm per inch of nominal valve size with an upper limit of the maximum allowable leakage in Table 3.4-1; and the measured leak rate for any given test cannot reduce the difference between the results of the previous test and the maximum allowable leakage specified in Table 3.4-1 by more than 50%:#

a. Every refueling outage during startup.

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve affecting the seating capability of the valve.

c. Following valve actuation due to automatic or manual action or flow through the valve for valves identified in Table 3.4-1 by an asterisk.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

To satisfy ALARA requirements, leakage may be measured indirectly (as from performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	DESCRIPTION	MAXIMUM ALLOWABLE LEAKAGE**
Q2E11V001A	12" GATE	5.000 GPM
Q2E11V001B	12" GATE	5.000 GPM
Q2E11V016A	12" GATE	5.000 GPM
Q2E11V016B	12" GATE	5.000 GPM
Q2E11V021A	6" CHECK	3.000 GPM
Q2E11V021B	6" CHECK	3.000 GPM
Q2E11V021C	6" CHECK	3.000 GPM
* Q2E21V032A	12" CHECK	5.000 GPM
* Q2E21V032B	12" CHECK	5.000 GPM
* Q2E21V032C	12" CHECK	5.000 GPM
* Q2E21V037A	12" CHECK	5.000 GPM
* Q2E21V037B	12" CHECK	5.000 GPM
* Q2E21V037C	12" CHECK	5.000 GPM
Q2E11V042A	10" CHECK	5.000 GPM
Q2E11V042B	10" CHECK	5.000 GPM
* Q2E21V076A	6" CHECK	3.000 GPM
* Q2E21V076B	6" CHECK	3.000 GPM
* Q2E21V077A	6" CHECK	3.000 GPM
* Q2E21V077B	6" CHECK	3.000 GPM
Q2E21V077C	6" CHECK	3.000 GPM

* Indicates the requirements of Section 4.4.7.2.2 Item (c) are applicable.

** The measured leak rate for any given test cannot reduce the difference between the results of the previous test and the maximum allowable leakage specified in Table 3.4-1 by more than 50%.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 41 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 reference 1 Alabama Power Company (the licensee) submitted proposed revisions to the Technical Specifications for Farley 1 and 2 reactor coolant system pressure isolation valves (PIV's). These proposed Technical Specifications reflected previous resolution of issues and staff concerns as outlined in references 1 and 2, as well as in previous correspondence. Our discussion and evaluation follows.

Discussion and Evaluation

As a result of the Event V Order issued for Farley 1 on April 20, 1981, the Technical Specifications required leak rate testing of only four PIV's valves. The acceptance criteria for valve leakage for these valves is as follows:

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable. However, for initial tests, or tests following valve repair or replacement, leakage rates less than or equal to 5.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm, are considered unacceptable.

With regard to the Farley 2 PIV test program, the NRC staff position originally was that acceptable leak rates for PIV's should not be greater than one gpm. However, the NRC staff had granted approval to use higher leak rate acceptance criteria, similar to Farley 1, at Farley 2 on two separate occasions, each on a one-time-only basis. (See references 3 and 5). The licensee proposes the higher acceptance criteria permanently at Farley 2 and uniformly for both Farley 1 and 2.

The licensee proposes to make the PIV leak test program identical at both Farley 1 and Farley 2. The PIV list for each plant will consist of 20 valves. The staff had previously determined (see Reference 2) that these valves constitute the PIV list for Farley 2. The staff concurs with this approach at Farley 1 as well, for the reasons noted in Reference 2.

The maximum allowable leak rate for each PIV is proposed to be 3 gpm for 6 inch valves and 5 gpm for the remaining valves which are either 10 inch or 12 inch. This is equivalent to an allowable leak rate of one-half gpm for each inch of valve size with maximum upper limit of 5 gpm. In addition the licensee proposes that the measured leak rate for any given test should not reduce the difference between the test results of a previous test and the maximum leak rate by more than 50 percent.

The staff concurs with the licensee's proposal. An approach similar to that advocated by the licensee is now being considered by the NRC staff, and if approved by NRC management will result in a Standard Technical Specification change. The change is justified as follows:

(a) The original one gpm criterion for Farley 2 was more-or-less arbitrary. This criterion has been imposed on all plants licensed since the TMI-2 accident. It was based on a very conservative estimate of the pressure relief system capacity for a plant. The one gpm criteria is not an indicator of imminent accelerated deterioration or potential valve failure.

(b) In a study which was sponsored by the staff (EGG Report EGG-NTAP-6175, February 1983, "In Service Leak Testing of Primary Pressure Isolation Valves", R. A. Livingston) it was concluded that allowable leak rates based on valve size were superior to a single allowable value because a single allowable value imposes an unjustified penalty on larger valves without providing information on potential valve degradation. Also, the larger valves must be repaired in-place which subjects plant personnel to radiation exposure in order to meet an overly conservative standard. In addition, an indexing criterion to account for gross increases in leakage from one test to a later test, as found in the ASME code, paragraph IWV-3427 (b) is a direct indicator of potential valve degradation. Since such an indexing criterion will be used by the licensee, this will provide at least as good, if not better, an indication of valve deterioration as the one gpm criterion.

(c) Previous safety evaluations (see attachments to references 3 and 5) in support of the one time Technical Specification changes allowed for Farley 2 provided analyses of data submitted by the licensee in support of his requests (see, for example, Reference 4). In support of the staff's position the following paragraph is quoted from Reference 5:

"Alabama Power Company (APCo) has supported their request by providing actual leakage data accumulated over approximately two years of leak testing these valves for Units 1 and 2 to the two different criteria. APCo provided the following historical data: The Unit 1 valves have been exposed to sixteen tests in past outages and resulted in six failures when the utility had arbitrarily imposed the Unit 2 1 gpm criteria. Personnel radiation exposure was estimated to be 25 rem to meet the 1 gpm criteria, but only 2.5 rem to meet the 1 to 5 gpm criterion. The utility also states that of the valves which failed the 1 gpm criterion and those that failed the 1 to 5 gpm criterion no discernible differences in seating surfaces could be found, and no evidence of impending valve failures were found in any of the valves that failed either criterion."

The staff's contemplated approach to monitoring leak rates for PIV's is to be found in ASME Code paragraph IWV-3427(b) of Section XI. The licensee's approach is somewhat more conservative than the NRC staff's proposal since it calls for immediate repair or replacement of valves which do not meet the "50% criterion." The staff's proposal would not immediately require repair or replacement unless the increase in leakage rate was pronounced. It is considered that the staff's approach is desirable since it allows some flexibility when the increase in leak rates is on the borderline of acceptability.

SAFETY SUMMARY

In conclusion, the PIV leak rate criteria proposed by the licensee is acceptable to the staff. This conclusion is based on an evaluation of the data submitted by the licensee and our independent staff study (EGG Report).

References:

1 Alabama Power Company letter to USNRC dated April 10, 1984, Farley 1 and 2, "Proposed Technical Specification change for Leakage Testing of Reactor Coolant System Pressure Isolation Valves"

2 USNRC letter to Alabama Power Company dated January 26, 1984, Farley 1 and 2, "Relief from ASME Section XI Requirement for Inservice Testing Program for Pumps and Valves"

3 USNRC letter to Alabama Power Company dated September 8, 1983, Farley 2, Amendment No. 25 to License NPF8

4 Alabama Power Company letter to USNRC dated June 3, 1983, Farley 2, "RCS Pressure Isolation Valve Leak Test Results"

5 USNRC letter to Alabama Power Company dated November 24, 1982, Farley 2, Amendment No. 20 to Facility Operating License No. NPF8

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: October 15, 1984

Principal Contributor:

O. Rothberg