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Docket No. 50-348

Mr. F. L. Clayton
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

Dear Mr. Clayton:

The Commission has issued the enclosed Amendment No. 25 to Facility Operating License No. NPF-2 for the Joseph M. Farley Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated September 25, 1981.

The amendment incorporates the TMI-2 Lessons Learned Category "A" Technical Specification changes. The Technical Specification changes are supported by the Safety Evaluation Report as transmitted to you by letter dated April 3, 1980.

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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A copy of the Notice of Issuance is also enclosed.

Sincerely,

Original signed by:
S. A. Varga

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

Enclosures:
1. Amendment No. 25 to NPF-2
2. Notice of Issuance

cc w/enclosures:
See next page

cp 9/29/81

OK - no legal objection to order or notice

OFFICE	ORB#1:DL	ORB#1:DL	ORB#1:DL	AD/OR:DL	OELD		
SURNAME	CParrish	EReeves:ds	SVarga	TNovak	DSwanson		
DATE	9/28/81	9/29/81	9/29/81	9/29/81	10/1/81		

Mr. F. L. Clayton
Alabama Power Company

cc: Mr. W. O. Whitt
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Columbia, Alabama 36319

State Department of Public Health
ATTN: State Health Officer
State Office Building
Montgomery, Alabama 36104

Regional Radiation Representatives
EPA Region IV
345 Courtland Street, N.E.
Atlanta, Georgia 30308

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. 25
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 September 25, 1981, 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, the provisions of the Act, and the rules and 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 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. 25 , 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 the date of its 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 5, 1981

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 25 TO FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Revise Appendix A as follows:

Remove Pages	Insert Pages
IV	IV
XI	XI
XVII	XVII
3/4 3-23	3/4 3-23
3/4 3-53	3/4 3-53
3/4 3-54	3/4 3-54
3/4 3-55	3/4 3-55
3/4 4-6	3/4 4-6
-------	3/4 4-6a
B3/4 3-4	B3/4 3-4
B3/4 4-2	B3/4 4-2
--------	B3/4 4-2a
6-1	6-1
6-4	6-4
6-5	6-5
--------	6-13a

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION	PAGE
3/4.3 INSTRUMENTATION	
3/4.3.1 PROTECTIVE INSTRUMENTATION	B 3/4 3-1
3/4.3.2 ENGINEERED SAFETY FEATURE INSTRUMENTATION	B 3/4 3-1
3/4.3.3 MONITORING INSTRUMENTATION	B 3/4 3-1
3/4.4 REACTOR COOLANT SYSTEM	
3/4.4.1 REACTOR COOLANT LOOPS	B 3/4 4-1
3/4.4.2 and 3/4.4.3 SAFETY VALVES	B 3/4 4-1
3/4.4.4 PRESSURIZER	B 3/4 4-2
3/4.4.4a RELIEF VALVES (PORV's)	B 3/4 4-2
3/4.4.5 STEAM GENERATORS	B 3/4 4-2
3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE	B 3/4 4-4
3/4.4.7 CHEMISTRY	B 3/4 4-5
3/4.4.8 SPECIFIC ACTIVITY	B 3/4 4-5
3/4.4.9 PRESSURE/TEMPERATURE LIMITS	B 3/4 4-6
3/4.4.10 STRUCTURAL INTEGRITY	B 3/4 4-11

ADMINISTRATIVE CONTROLS

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	≤ 4.0 psig	≤ 4.5 psig
d. Pressurizer Pressure--Low	≥ 1850 psig	≥ 1840 psig
e. Differential Pressure Between Steam Lines--High	≤ 100 psi	≤ 112 psi
f. Steam Line Pressure--Low	≥ 585 psig steam line pressure	≥ 575 psig steam line pressure

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of channels shown in Table 3.3-11, restore the inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11; restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Coolant Outlet Temperature-T_{Hot}-Wide Range	2	1
2. Reactor Coolant Inlet Temperature-T_{Cold}-Wide Range	2	1
3. Reactor Coolant Pressure-Wide Range	2	1
4. Steam Generator Water Level-Wide Range or Narrow Range	2/steam generator	1/steam generator
5. Refueling Water Storage Tank Water Level	2	1
6. Containment Pressure	2	1
7. Pressurizer Water Level	2	1
8. Steam Line Pressure	2/steam generator	1/steam generator
#9. Auxiliary Feedwater Flow Rate	2	1
#10. Reactor Coolant System Subcooling Margin Monitor	2	1
#*11. PORV Position Indicator	1/valve	1/valve
#**12. PORV Block Valve Position Indicator	1/valve	1/valve
# 13. Safety Valve Position Indication (One channel is position indicator and one channel is discharge temperature)	2/valve	1/valve

*Not applicable if the associated block valve is in the closed position.
**Not applicable if the block valve is verified in the closed position and power removed.

Specification effective 90 days following the return to power following the third refueling outage.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Coolant Outlet Temperature-T_{Hot}-Wide Range	M	R
2. Reactor Coolant Temperature-T_{Cold}-Wide Range	M	R
3. Reactor Coolant Pressure-Wide Range	M	R
4. Steam Generator Water Level- Wide Range or Narrow Range	M	R
5. Refueling Water Storage Tank Water Level	M	R
6. Containment Pressure	M	R
7. Pressurizer Water Level	M	R
8. Steam Line Pressure	M	R
# 9. Auxiliary Feedwater Flow Rate	M	R
#10. Reactor Coolant System Subcooling Margin Monitor	M	R
#*11. PORV Position Indicator	M	R
#**12. PORV Block Valve Position Indicator	M	R
#13. Safety Valve Position Indicator	M	R

*Not applicable if the associated block valve is in the closed position.
**Not applicable if the block valve is verified in the closed position and power removed.
Specification effective 90 days following the return to power following the third refueling outage.

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with at least 125 kw of pressurizer heaters and a water volume of less than or equal to 868 (63.5% indicated) cubic feet.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.4.2 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters.

4.4.4.3 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring circuit current at least once per 92 days.

---

*Limit not applicable during either a THERMAL POWER ramp change in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step change in excess of 10% of RATED THERMAL POWER.

Specification effective 90 days following the return to power following the third refueling outage.

3/4.4.4.a RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4.a Two power relief valves (PORV's) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.a.1 Each PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION and operating the valve through one cycle of full travel.

*4.4.4.a.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with the power removed in order to meet the ACTION requirements of a. above.

#Specification effective 90 days following the return to power following the third refueling outage.

*Implementation of this specification is deferred until resolution of the Unit 1 Technical Specification upgrade.

3/4.3.3.7 HIGH ENERGY LINE BREAK ISOLATION SENSORS

The high energy line break isolation sensors are designed to mitigate the consequences of the discharge of steam and/or water to the affected room and other lines and systems contained therein. In addition, the sensors will initiate signals that will alert the operator to bring the plant to a shutdown condition.

3/4.3.3.8 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available for selected plant parameters to monitor and assess these variables following an accident.

3/4.3.3.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.4.2 and 3/4.4.3 SAFETY VALVES (Continued)

than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves lift setting will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE assures that the plant will be able to establish natural circulation.

3/4.4.4.a RELIEF VALVES (PORV'S)

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operaton of the PORV's minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for Inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, 3/4.4.5 STEAM GENERATORS (Continued)

manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. All CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. A Fire Brigade of at least 5 members shall be maintained onsite at all times. The Fire Brigade shall not include 3 members of the minimum shift crew necessary for safe shutdown of the unit or any personnel required for other essential functions during a fire emergency.

6.2.3 SHIFT TECHNICAL ADVISOR

6.2.3.1 The Shift Technical Advisor shall serve in an advisory capacity to the Shift Supervisor primarily in the assessment of accident and transient occurrences.

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL	1	1*
OL	2	1
Non-Licensed	2	1
STA	1[a]	None

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS.

#Shift crew composition (including an individual qualified in radiation protection procedures) may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

a/ Individual may fill the same position on Unit 2.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, except for (1) the Chemistry and Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976, except for Fire Brigade training sessions which shall be held at least quarterly.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Plant Manager
Vice Chairman:	Assistant Plant Manager
Member:	Technical Superintendent
Member:	Operations Superintendent
Member: (Non-voting)	Plant Quality Assurance Engineer
Member:	Maintenance Superintendent

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than one alternate shall participate as voting members in PORC activities at any one time.

*The Minimum qualifications requirement for the Chemistry and Health Physics Supervisor shall become effective when the initial incumbent in this position is replaced.

6.8.3 The following programs shall be maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include recirculation portions of the containment spray, safety injection and chemical and volume control systems, the waste gas system, the Reactor Coolant sampling system, the residual heat removal system, and the containment atmosphere sampling system. The program shall include the following.

(i) Preventative maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system with the exception of the waste gas system and the containment atmosphere sampling system which are "snoop" tested at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in certain plant areas where personnel may be present under accident conditions. This program shall include the following:

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analyses equipment.

c. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System Subcooling margin. This program shall include the training of personnel and the procedures for monitoring.

#Specification effective 90 days following the return to power following the third refueling outage.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-348

ALABAMA POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 25 to Facility Operating License No. NPF-2 issued to Alabama Power Company (the licensee), which revised Technical Specifications for operation of the Joseph M. Farley Nuclear Plant, Unit No. 1 (the facility) located in Houston County, Alabama. The amendment is effective as of the date of issuance.

The amendment incorporates the TMI-2 Lessons Learned Category "A" Technical Specification changes.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since this amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated September 25, 1981, (2) Amendment No. 25 to License No. NPF-2, and (3) the Commission's letter dated October 5, 1981 . All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the George S. Houston Memorial Library, 212 W. Burdeshaw Street, Dothan, Alabama 36303. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention, Director, Division of Licensing.

Dated at Bethesda, Maryland, this 5th day of October, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

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