JULY 3 1 1979

Docket No. 50-348

Distribution
Docket File 50-348
NRC PDR
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B. Harless
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R. Diggs (TAC 8435)
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ACRS (16)
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J. Buchanan
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W. Russell
G. Zech

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

Dear Mr. Barton:

The Commission has issued the enclosed Amendment No. 13 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 6, 1978 (superceding your application of March 17, 1977) supplemented by letters dated November 3, 9, and 17, 1978, and January 4, March 21 and April 17, 1979.

The amendment approves the permanent overpressure mitigating system and associated Technical Specifications. The system will minimize the potential for water-solid overpressurization. Minor changes were made to your proposed Technical Specifications. These changes have been discussed with your staff who agreed to our revisions.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,



Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

CP
Side 1

Enclosures:
1. Amendment No. 13 to NPF-1
2. Safety Evaluation
3. Notice of Issuance

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cc: w/enclosures
See next page *SEE PREVIOUS YELLOW FOR CONCURRENCE

TAC 8435

concurrence limited to FR notice + letter

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| SURNAME | EAReeves:jb | GGZech * | CSParrish* | DBrinkman* | TPSpeis | ASchwencer |
| DATE | 07/31/79 | 07/ /79 | 07/ /79 | 07/ /79 | 07/26/79 | 07/31/79 |

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Distribution

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See next page

TAC 8435

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| SURNAME | EAReeves:jb | GGZech | CSParrish | DBrinkman | ~~RIsrael~~ TPSpeis | AScwhencer | WPGammill | |
| DATE | 07/ /79 | 07/19/79 | 07/20/79 | 07/20/79 | 07/ /79 | 07/ /79 | 07/ /79 | 07/ /79 |

NRC FORM 318 (9-76) NRCM 0240 ★ U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

NRC PDR

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

July 31, 1979

Docket No. 50-348

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

Dear Mr. Barton:

The Commission has issued the enclosed Amendment No. 13 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 6, 1978 (superceding your application of March 17, 1977) supplemented by letters dated November 3, 9, and 17, 1978, and January 4, March 21 and April 17, 1979.

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Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 13 to NPF-1
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

7908220531

cc: Ruble A. Thomas, Vice President
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Birmingham, Alabama 35202

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

Attorney General
State Capitol
Montgomery, Alabama 36104

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
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. 13
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 6, 1978 (superceding your application of March 17, 1977) supplemented by letters dated November 3, 9, 17, 1978 and January 4, March 21, and April 17, 1979, 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. 13 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Delete license condition 2.C.(3)(b).

4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: July 31, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 13

FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. Revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Corresponding overleaf pages are also provided to maintain document completeness.

Pages

V
3/4 1-9
3/4 4-2
3/4 4-3
3/4 4-30 (added)
3/4 4-31 (added)
3/4 4-32 (was 3/4 4-30)
3/4 4-33 (was 3/4 4-31)
B 3/4 1-3
B 3/4 1-4 (added)
B 3/4 4-1
B 3/4 4-2
B 3/4 4-3
B 3/4 4-11

## INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

## CHARGING PUMP - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5* and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump is restored to OPERABLE status.

### SURVEILLANCE REQUIREMENTS

4.1.2.3.1 At least the above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a discharge pressure of $\geq$ 2458 psig when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours by verifying that the motor cirucit breakers have been removed from their electrical power supply circuits.

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is $\leq$ 180°F.

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1. All reactor coolant loops shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.*

ACTION:

Above P-7, comply with either of the following ACTIONS:

a. With one reactor coolant loop and associated pump not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to less than 36% of RATED THERMAL POWER and the following ESF instrumentation channels associated with the loop not in operation, are placed in their tripped condition within 1 hour:

   1. $T_{avg}$ -- Low-Low channel used in the coincidence circuit with Steam Flow - High for Steam Line Isolation.

   2. Steam Line Pressure - Low for Safety Injection.

   3. Steam Flow-High Channel used for MSIV Isolation.

   4. Differential Pressure Between Steam Lines - High channel used for Safety Injection (trip all bistables which indicate low active loop steam pressure with respect to the idle loop steam pressure).

b. With one reactor coolant loop and associated pump not in operation, subsequent STARTUP and POWER OPERATION above 36% of RATED THERMAL POWER may proceed provided:

   1. The following actions have been completed with the reactor in at least HOT STANDBY.

      a) Reduce the overtemperature ΔT trip setpoint to the value specified in Specification 2.2.1 for 2 loop operation.

*See Special Test Exception 3.10.4.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

b. With $K_{eff}$ < 1.0, operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant or residual heat removal pump.*

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1. With one reactor coolant loop and associated pump not in operation, at least once per 31 days determine that:

a. The applicable reactor trip system and/or ESF actuation system instrumentation channels specified in the ACTION statements above have been placed in their tripped conditions, and

b. The P-8 interlock setpoint is within the following limits if the P-8 interlock was reset for 2 loop operation $\leq$ 66% of RATED THERMAL POWER.

*All reactor coolant pumps and residual heat removal pumps may be deenergized for up to one (1) hour provided no operations are permitted which could cause dilution of reactor coolant system boron concentration.

## LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum cooldown of 200°F in any one hour period,

b. A maximum heatup of 100°F in any one hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

## SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per hour during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

## SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each RHR relief valve shall be demonstrated OPERABLE by:

a. Verifying the RHR relief valve isolation valves (8701a, 8701b, 8702a and 8702b) are open at least once per 72 hours when the RHR relief valve is being used for overpressure protection.

b. Testing in accordance with the inservice test requirements for ASME Category C valves pursuant to Specification 4.0.5.

c. Verification of the RHR relief valve setpoint, of at least one RHR relief valve, at least once per 18 months on a rotating basis.

4.4.9.3.2 The RCS vent shall be verified to be open at least once per 12 hours* when the vent is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

## SURVEILLANCE REQUIREMENTS

a. Per the requirements of Specification 4.0.5 and

b. By the augmented program specified in Specifications 4.4.10.2 and 4.4.10.3

4.4.10.2 In addition to the requirements of Specification 4.0.5, the Reactor Coolant Pump flywheels shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14 Revision 1, August 1975.

4.4.10.3 In addition to the requirements of Specification 4.0.5 the three main steamlines from the rigid anchor points of the containment penetrations downstream to and including the main steam header shall be inspected. The extent of the inservice examinations completed during each inspection interval (IWA 2400, ASME Code, 1974 Edition, Section XI) shall provide 100 percent volumetric examination of circumferential and longitudinal pipe welds to the extent practical. The areas subject to examination are those defined in accordance with examination category C-G for Class 2 piping welds in Table IWC-2520.

## 3/4.1.2 BORATION SYSTEMS (Continued)

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 180°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single RHR relief valve.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2000 gallons of 7000 ppm borated water from the boric acid storage tanks or 9,000 gallons of 2000 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

## 3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment and associated accident analysis. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. In addition, those accidents analyses affected by a misaligned rod are re-evaluated to confirm that the results remain valid during future operation.

# 3/4.4 REACTOR COOLANT SYSTEM

## BASES

### 3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. With one reactor coolant loop not in operation, THERMAL POWER is restricted to < 36 percent of RATED THERMAL POWER until the Overtemperature $\Delta T$ trip is reset. Either action ensures that the DNBR will be maintained above 1.30. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (36 percent of RATED THERMAL POWER).

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a RHR loop into operation in the shutdown cooling mode if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs $\leq 310^{\circ}F$ are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which would exceed the limits of Appendix G to 10CFR Part 50. The RCS will be protected against overpressurization transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting from starting the RCP's to when the secondary water temperature of each steam generator is less than $50^{\circ}F$ above each of the RCS cold leg temperatures.

### 3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater

STEAM GENERATORS (Continued)

limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% degradation (60% tube wall thickness). Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specification, if necessary.

vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the $\Delta RT_{NDT}$ determined from the surveillance capsule is different from the calculated $\Delta RT_{NDT}$ for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two RHR relief valves or an RCS vent opening of $\geq$ 2.85 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are $\leq$ 310°F. Either RHR relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator $\leq$ 50°F above the RCS cold leg temperatures, or (2) the start of 3 charging pumps and their injection into a water solid RCS.

## 3/4.4.10 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# SUPPORTING AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE NO. NPF-2

# ALABAMA POWER COMPANY

# JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 1

# DOCKET NO. 50-348

## Introduction

By letter to Alabama Power Company (APC) dated December 29, 1976, the NRC requested an evaluation of the Farley Nuclear Plant (FNP) system designs to determine susceptibility to overpressurization events. We also requested an analysis of the possible events, licensee proposed interim and permanent systems, and procedure modifications to reduce the likelihood and consequences of such events. By letter dated March 17, 1977 APC proposed development and installation of an Overpressure Mitigating System (OPMS) using the existing pressurizer power operated relief valves as a long term solution to the problem. However, by letter of September 6, 1978 APC modified its earlier proposal. APC's new proposal would use the existing pressure relieving capacity of the residual heat removal (RHR) system relief valves. Subsequently, by letters dated November 3, 9 and 17, 1978 and January 4, March 21, and April 17, 1979, APC provided information on the OPMS design and transient analyses, proposed Technical Specifications for OPMS operability and surveillance, and responded to our specific concerns. The OPMS, as now designed, is based upon operation of passive RHR relief valves without additional instrumentation (except alarms) or relays, solenoids and valve operators. The system is designed to prevent a reactor coolant system (RCS) transient from exceeding the pressure and temperature limits of the Technical Specifications for Farley Nuclear Plant (FNP), as required by Appendix G to 10 CFR 50.

## Background Discussion

Over the last few years, incidents identified as pressure transients have occurred in pressurized water reactors (PWRs). As used in this report "pressure transients" refers to events during which the temperature-pressure limits of the reactor vessel, as shown in the Technical Specifications, are exceeded. All of these incidents occurred at relatively low temperatures (less than 200°F) where the reactor vessel material toughness (resistance to brittle failure) is reduced from that which exists at normal operating temperatures.

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The "Technical Report on Reactor Vessel Pressure Transients" in NUREG-0138 summarizes the technical considerations relevant to this matter, discusses the safety concerns and existing safety margins of operating reactors, and describes the regulatory actions taken to resolve this issue by reducing the likelihood of future pressure transient events at operating reactors. A brief discussion is presented here.

Reactor vessels are constructed in accordance with the ASME Boiler and Pressure Vessel Code. Steels used are particularly tough at normal reactor operating pressure and temperature conditions. However, these steels are less tough if subject to relatively high pressures at relatively low temperatures. Thus, restrictions are placed on the system pressure during startup and shutdown operations.

At operating (hot) temperatures, the pressures allowed by Appendix G limits are above the setpoint of the installed pressurizer code safety relief valves. However, most operating PWRs did not have automatic pressure relief device setpoints low enough to mitigate pressure transients during cold conditions (startup and shutdown) when the RCS is water-solid and non-vented.

By letter dated December 29, 1976 prior to issuing an operating license for FNP, we requested the applicant to begin efforts to design and install plant systems to mitigate the consequences of pressure transients at low temperatures. We also requested that operating procedures be examined and administrative changes be made to prevent initiating overpressure events. We also concluded that interim administrative controls should be imposed to assure safe operation until permanent overpressure mitigating hardware could be installed.

By letters dated January 24, March 17, and May 24, 1977 (in response to our May 3, 1977 letter) APC provided preliminary information describing interim measures to prevent these transients. The applicant proposed several modifications to administrative procedures, design, and operator training as discussed below:

1. Operator Training: During cold license training, operators would be briefed on the types of events that could cause overpressurization based on changes made in the procedures to minimize the probability of such events.

2. RHR Relief Valve: The setpoints of the two RHR relief valves would be set at 450 psig which is significantly below the isolation pressure of the RHR system. One valve has a 900 gpm relief capacity which is above the flow capacity of a single charging pump.

3. Steam Bubble: A steam bubble would be formed in the pressurizer at 160°F when the plant was being heated up. The bubble would be collapsed at 160°F when the plant is cooled down. This procedure would minimize the amount of time in a water-solid condition.

4. Charging Pump: Only one charging pump would be operable at RCS temperatures below 200°F. This would limit the potential volumetric insurge. With an open path to the RHR relief valve, no overpressurization would occur if the RCS letdown line were inadvertently closed. The other charging pumps would have power removed.

5. Letdown Line: The RCS letdown heat exchanger control valve would be placed in manual control prior to starting or stopping an RHR pump when the RCS is in a water-solid condition. This would preclude a spurious isolation of the RHR system.

6. Reactor Coolant Pumps: The procedures would include a precaution to verify that RHR suction valves were open prior to starting an RCS pump during water-solid operation. Also, included would be procedure limitations while in water-solid conditions on starting an RCS pump after the loss of RCS flow when seal injection temperature was less than RCS temperature. This would require either (a) the start of an RCS pump within 5 minutes or (b) the establishment of a steam bubble in the pressurizer or (c) a reduction of RCS pressure to 50-100 psig and the securing of seal flow for at least two hours prior to re-establishing seal flow and starting an RCS pump.

7. Accumulators: The accumulator isolation valves would be closed and the power locked-out whenever the plant is on the RHR cooling mode.

8. Alarm: An alarm utilizing existing RCS wide-range pressure instrumentati would be installed. This alarm would annunciate on the Main Control Board (audio and visual) when the RCS pressure and temperature approach Appendix G limits.

We found the proposed administrative and design changes acceptable as interim measures to minimize the likelihood of a water-solid overpressurization pending confirmation by the Inspection and Enforcement Office of NRC of their implementation. Because of the minimal neutron damage to be suffered by the pressure vessel during its first operating cycle, we concluded that no credible event would cause vessel rupture due to overpressurization during this period. Because of APC's proposed administrative procedures and the pressure vessel fracture toughness, we concluded that the reactor could operate for its first cycle with reasonable assurance that the health and safety of the public are protected.

APC became a member of the utility group in developing long-term solutions to mitigate the consequences of RCS pressure transients during water-solid operation. The long-term design modifications considered by the group would use either the power-operated relief valves or RHR system relief valves to preclude violating Appendix G limits. We required in license condition 2.C.(3)(b) that an effective overpressure protection system be installed prior to the initiation of the second operating cycle. This has been accomplished.

The purpose of this safety evaluation is to document the basis for our approval of APC's OPMS design, their analyses, and the issuance by NRC of related Technical Specifications.

Evaluation

The proposed overall approach to eliminating overpressure events incorporates administrative, procedural and hardware controls with reliance upon the plant operator for the principal line of defense. Preventive administrative and procedural measures include: (1) procedural precautions, (2) de-energization of components during cold shutdown, (3) avoidance of water solid reactor coolant system whenever possible, and (4) addition of an overpressure protection system which incorporates low pressure relief using the existing RHR system relief valves.

Water-solid overpressure mitigation systems are designed to mitigate the consequences of an overpressure event in the RCS when at low temperatures. The system at each plant must be designed to prevent violation of 10 CFR 50 Appendix G limits. In particular, it must:

1. Perform its function assuming any single active component failure,

2. For the worst mass and heat input events postulated, not violate Appendix G limits, as demonstrated through appropriate calculational techniques,

3. Meet IEE-279 requirements and provide overpressure alarm,

4. Provide the ability to be tested to assure its operability,

5. Function during and after an Operating Basis Earthquake (OBE), and

6. Not require offsite power for proper operation.

The OPMS proposed by APC in letter dated September 6, 1978, as supplemented, is an integral part of the RHR system relying on the RHR pump suction line relief valves to provide pressure relief. The RHR relief valves (RHRRV) are spring loaded, bellows type valves which have a setpoint of 450 psig. The valves will be fully open at 495 psig (i.e., they have 10% accumulation). There are two isolation valves between each of the RHRRVs and the RCS. The RHR system suction lines are automatically isolated when the RCS pressure exceeds 700 psig. The RHR system suction valves inside containment will be opened whenever the RCS temperature is 310°F or lower, thereby aligning the RHR relief valves for RCS pressure protection.

APC provided a single failure analysis which demonstrates that no single active failure can disable the mitigation system such that Appendix G limits are violated. Single failures considered included electrical faults, operator action, suction valve failure, relief valve failure, and loss of offsite power.

APC provided Appendix G curves for 100°F/hr cooldown and heatup rates (applicable for 8 effective full power years) for comparison against potential overpressure event maximum pressures. APC used the modified Westinghouse generic water-solid overpressure model with the specific RCS volume for the FNP and with a conservative steam generator heat transfer area. The worst mass input event was assumed to be the inadvertent operation of three high head safety injection pumps with a maximum total flow rate of 1000 gallons per minute at zero psig backpressure. The worst heat input event was assumed to be the starting of a single reactor coolant pump with a temperature differential of 50°F existing between the RCS and the steam generator. The maximum calculated RCS pressures for these postulated worst mass and heat input events remained below the pressures allowed by the Appendix G curves for transients initiated below 310°F and below 450 psig. For transients above 310°F, the pressurizer safety valves would relieve pressure to prevent violation of Appendix G limits. However, there is a hypothetical condition where it appears that the Appendix G curve might be exceeded by a pressure transient. The postulated case is one at a cooldown rate between 60°F/hr and 100°F/hr. For cooldown rates in this range while at low RCS temperatures (below 126°F), starting one charging pump could result in violating the 60°F/hr or 100°/hr Appendix G curves. This condition is highly improbable as cooldown rates of 60°F/hr to 100°F/hr at such low temperatures are very unlikely if not impossible to achieve. For these reasons the staff finds the maximum calculated water-solid overpressure event pressure analyses and consequences to be acceptable.

The RHR relief valves, which are installed on the two RHR pump suction lines, discharge to the pressurizer relief tank. The FNP analysis conservatively takes no credit for RHR relief valve flow until 10 percent overpressure (495 psig) has been achieved at which point the valve was assumed to be instantaneously and fully open. Valve backpressure was reviewed to assure that flow degradation due to flashing was included in the analysis. The RHR valves are spring-loaded and have no electrical components. The auto-closure interlock and the open permissive control circuits of the motor-operated RHR isolation valves meet the requirements of IEEE 279-1971. Power supplies for the RHR isolation valves, the pressurizer pressure sensors, and the RCS temperature sensors are designed so that no single failure in the electrical system or the loss of offsite power would isolate both of the RHR relief valves. Thus, we find the RHR relief valves, isolation valve circuitry and power supplies acceptable.

APC has committed to test the two RHR relief valves on an accelerated basis from that required by the ASME code. Bench tests will be done at 18 month intervals on a rotating basis for at least one of the RHR relief valves to check the setpoint. The Technical Specifications surveillance has been modified accordingly. The RHR relief valves are certified by the manufacturer to be capable of withstanding an OBE with no degradation of performance. The RHR system piping from the RCS hot leg to the RHR relief valves is quality group A per 10 CFR 50.55(a). We find the valve testing and certified ability of the relief valves to operate during and after an OBE acceptable.

Changes to FNP Technical Specifications (Limiting Conditions for Operation) submitted by APC (with minor changes by our staff) would ensure proper operation of the system when required. With the modifications we find these changes acceptable. These modified changes are listed below:

1. <u>Specification 3.4.9.3</u>: The RHR suction valves must be open and the RHR relief valves must be operable or a RCS vent must be open whenever the RCS temperature is less than or equal to 310°F.

2. <u>Specification 3.4.1 (Below P-7)</u>: A reactor coolant pump cannot be started when the RCS temperature is below 310°F unless the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures, or the pressurizer water volume is less than 24 percent level.

APC has installed a permanent alarm in the control room for Unit No. 1 (and will install one prior to initial fuel loading for Unit No. 2) to alert the operator if the RHR isolation valves are not fully open when the RCS temperature is less than or equal to 300°F. This is a seismic Category I alarm designed to the requirements of IEEE-279 up to the annunciator light. The alarm has an unshared annunciator window. APC has installed another alarm which would alert the operator to the existence of an overpressure event if the RCS pressure exceeds 450 psig.

We find the proposed overpressure mitigation system for the Joseph M. Farley Nuclear Plant, Unit No. 1 to be acceptable because it satisfies our requirements noted above. Although this evaluation has been prepared in connection with a pending licensing action on Farley Unit No. 1, it applies to the identified OPMS being installed in Farley Unit No. 2. This evaluation may be cited by the NRC as the basis for approval of the Unit No. 2 OPMS at a future date.

## Environmental Consideration

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.

## Conclusion

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.

Date: July 31, 1979

7590-01

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. 13 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 approves the permanent overpressure mitigating system and associated changes to the Technical Specifications. The system will further minimize the potential for water-solid overpressurization of the reactor coolant system pressure boundary.

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 6, 1978 (superceding application of March 17, 1977), supplemented by letters dated November 3, 9, and 17, 1978, and January 4, March 21, and April 17, 1979, (2) Amendment No. 13 to License No. NPF-2, and (3) the Commission's related Safety Evaluation. 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 Operating Reactors.

Dated at Bethesda, Maryland, this 31st day of July, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors