

April 16, 1986

Docket Nos. 50-348
and 50-364

Mr. R. P. McDonald
Senior Vice President
Alabama Power Company
Post Office Box 2641
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Dear Mr. McDonald:

The Commission has issued the enclosed Amendment No. 63 to Facility Operating License No. NPF-2 and Amendment No. 54 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 November 27, 1985.

The amendments modify Technical Specifications (TS) to delete TS 3.4.9 Action Statement a. which required a plant shutdown when coolant activity exceeded a cumulative spiking limit. In addition, reporting requirements are changed from a 30-day report to an annual report. TS 1.9 Dose Equivalent Iodine I-131 is revised to reference Regulatory Guide (RG) 1.109. These changes are consistent with Generic Letter 85-19 and the cited RG.

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

Sincerely,

/s/
Edward A. Reeves, Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 63 to NPF-2
2. Amendment No. 54 to NPF-8
3. Safety Evaluation

cc: w/enclosures
See next page

LA: PAD#2
D Miller
3/27/86

PM: PAD#2
E Reeves:hc
4/1/86

D: PAD#2
L Rubenstein
4/1/86

OELD
B. A. Decker
4/19/86

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PDR ADOCK 05000348
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Mr. R. P. McDonald
Alabama Power Company

Joseph M. Farley Nuclear Plant

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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. 63
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 November 27, 1985, 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:

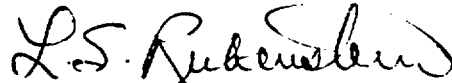
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P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 63, 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



Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 16, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 63

TO FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages

1-2
3/4 4-23
B3/4 4-5

Insert Pages

1-2
3/4 4-23
B3/4 4-5

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

AXIAL FLUX DIFFERENCE

1.2 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.3 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.4 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.5 A CHANNEL FUNCTIONAL TEST shall be:

- a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.
- b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

DEFINITIONS

CONTAINMENT INTEGRITY

1.6 CONTAINMENT INTEGRITY shall exist when:

- a. All penetrations required to be closed during accident conditions are either:
 - 1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or
 - 2) Closed by manual valves, blind flanges or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3,
- b. All equipment hatches are closed and sealed,
- c. Each air lock is OPERABLE pursuant to Specification 3.6.1.3,
- d. The containment leakage rates are within the limits of Specification 3.6.1.2, and
- e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 of Regulatory Guide 1.109, Revision 1, 1977.

REACTOR COOLANT SYSTEM

3/4.4.9 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.9 The specific activity of the primary coolant shall be limited to:

- a. Less than or equal to 1.0 microCurie per gram DOSE EQUIVALENT I-131, and
- b. Less than or equal to $100/\bar{E}$ microCurie per gram.

APPLICABILITY: MODES 1, 2, 3, 4 and 5

ACTION:

MODES 1, 2 and 3*:

- a. With the specific activity of the primary coolant greater than 1.0 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.
- b. With the specific activity of the primary coolant greater than $100/\bar{E}$ microCurie per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

*With T_{avg} greater than or equal to 500°F.

REACTOR COOLANT SYSTEM

ACTION: (Continued)

MODES 1, 2, 3, 4 and 5:

- a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/E microcuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.9 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

REACTOR COOLANT SYSTEM

BASES

3/4.4.8 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.9 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Farley site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microCuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

REACTOR COOLANT SYSTEM

BASES

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

- 1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3.
 - a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.
 - b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.
- 2) These limit lines shall be calculated periodically using methods provided below.
- 3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.



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. 54
License No. NPF-8

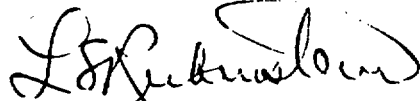
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Alabama Power Company (the licensee) dated November 27, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 54, 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



Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 16, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 54

TO FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages

1-2
3/4 4-23
B3/4 4-5

Insert Pages

1-2
3/4 4-23
B3/4 4-5

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

AXIAL FLUX DIFFERENCE

1.2 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.3 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.4 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.5 A CHANNEL FUNCTIONAL TEST shall be:

- a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.
- b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

DEFINITIONS

CONTAINMENT INTEGRITY

1.6 CONTAINMENT INTEGRITY shall exist when:

- a. All penetrations required to be closed during accident conditions are either:
 - 1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or
 - 2) Closed by manual valves, blind flanges or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3,
- b. All equipment hatches are closed and sealed,
- c. Each air lock is OPERABLE pursuant to Specification 3.6.1.3,
- d. The containment leakage rates are within the limits of Specification 3.6.1.2, and
- e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table E-7 of Regulatory Guide 1.109, Revision 1, 1977.

REACTOR COOLANT SYSTEM

3/4.4.9 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.9 The specific activity of the primary coolant shall be limited to:

- a. Less than or equal to 1.0 microCurie per gram DOSE EQUIVALENT I-131, and
- b. Less than or equal to $100/\bar{E}$ microCurie per gram.

APPLICABILITY: MODES 1, 2, 3, 4 and 5

ACTION:

MODES 1, 2 and 3*:

- a. With the specific activity of the primary coolant greater than 1.0 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.
- b. With the specific activity of the primary coolant greater than $100/\bar{E}$ microCurie per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

*With T_{avg} greater than or equal to 500°F.

REACTOR COOLANT SYSTEM

ACTION: (Continued)

MODES 1, 2, 3, 4 and 5:

- a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/E microcuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.9 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

REACTOR COOLANT SYSTEM

BASES

3/4.4.8 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.9 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Farley site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microCuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

REACTOR COOLANT SYSTEM

BASES

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.10 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

- 1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the first full-power service period.
 - a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.
 - b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.
- 2) These limit lines shall be calculated periodically using methods provided below.
- 3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. NPF-2
AND AMENDMENT NO. 54 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 and Evaluation

Generic Letter 85-19, Reporting Requirements on Primary Coolant Iodine Spikes, is part of an ongoing Commission effort to delete unnecessary reporting requirements. The NRC staff determined that reporting requirements related to primary coolant specific activity levels, specifically iodine spikes, can be reduced from a short term report to an item to be included in the annual report. Also, the staff determined that existing shutdown requirements based on exceeding the primary coolant iodine activity limits for an accumulated period of over 800 hours are no longer necessary. This is based on an improvement in the quality of nuclear fuel over the past 10 years since appropriate licensee actions would be initiated long before approaching the limit as currently specified.

By Generic Letter 85-19, the NRC staff reported these findings and provided model Technical Specifications (TS) in Standard Technical Specification format to all licensees for consideration. The model TS deleted Action Statement a. which required a plant shutdown when coolant activity exceeded a cumulative spiking limit. Alabama Power Company (APCo) proposed amending the Farley Units 1 and 2 TS's to match the model TS which are acceptable. APCo also included in the amendment request a proposal to change the definition of Dose Equivalent Iodine 131 (DEI-131) from one compatible with the TID 14844 dose conversion factors (DCF's) to the newer, less restrictive (i.e., results in lower doses) Regulatory Guide 1.109 DCF's. This change would bring the DEI definition used by APCo in line with that used in Generic Letter 85-19. Since the Farley Final Safety Analysis Report doses, still based on TID 14844 DCF's, would remain bounding, no increase in offsite consequences would result from this change.

Safety Summary

Because there will be no change in offsite consequences and the proposed amendment is consistent with NRC staff guidance contained in Generic Letter 85-19 and Regulatory Guide 1.109, we find the proposed Technical Specifications acceptable.

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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 meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). These amendments also involve changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, with respect to these items, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 16, 1986

Principal Contributor:

H. E. Gilpin