April 24, 1990

Docket No. 50-416

DISTRIBUTION
See attached sheet

Mr. W. T. Cottle
Vice President, Nuclear Operations
System Energy Resources, Inc.
Post Office Box 756
Port Gibson, Mississippi 39150

Dear Mr. Cottle:

SUBJECT: ISSUANCE OF AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1, REGARDING REACTOR PROTECTION SYSTEM (TAC NO. 68692)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 67 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated June 30, 1988, as supplemented by letter dated February 19, 1990.

The amendment changes the TS by increasing the surveillance test intervals and the allowable out-of-service times for the reactor protection system.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Original Signed By:

Lester L. Kintner, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 67 to NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PD21:DRPR	PM:PD21:DRPR	D:PD21:DRPR			
NAME	PAnderson	LKintner:bld	EAdensam			
DATE	4/23/90	4/23/90	4/24/90			

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Document Name: GGNS AMEND 68692

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AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF

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

SYSTEM ENERGY RESOURCES, INC., et al.

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 67
License No. NPF-29

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

   A. The application for amendment by System Energy Resources, Inc., (the licensee), dated June 30, 1988, as supplemented by letter dated February 19, 1990, 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.

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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-29 is hereby amended to read as follows:

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

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 67 , are hereby incorporated into this license. System Energy Resources, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By:

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 24, 1990

OFC	LA:PD21:DRPR	PM:PD21:DRPR	BC:SICB:DEST	OGC	D:PD21:DRPR	
NAME	PAnderson	LKintner:bld	SNewberry	RBachmann	EAdensam	
DATE	2/13/90	3/1/90	3/12/90	4/13/90	4/23/90	

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ATTACHMENT TO LICENSE AMENDMENT NO. 67

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3/4 3-1	3/4 3-1
3/4 3-5	3/4 3-5
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
B3/4 3-1	B3/4 3-1
B3/4 3-2	B3/4 3-2

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel and/or that trip system in the tripped condition* within twelve hours. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

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*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* per Specification 3.9.2 and shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when DRYWELL INTEGRITY is not required.

(g) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(h) This function shall be automatically bypassed when operating below the appropriate turbine first stage pressure setpoint of:

 (1) $\leq$ 26.9%** of the value of turbine first-stage pressure at valves wide open (VWO) steam flow when operating with rated feedwater temperature of greater than or equal to 420°F, or

 (2) $\leq$ 22.5%** of the value of turbine first-stage pressure at VWO steam flow when operating with rated feedwater temperature between 370°F and 420°F.

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*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**Allowable setpoint values of turbine first-stage pressure equivalent to THERMAL POWER less than 40% of RATED THERMAL POWER.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,[b]	S/U, W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:[f]				
a. Neutron Flux - High, Setdown	S/U,S,[b]	S/U, W	SA	2
	S	W	SA	3, 5
b. Flow Biased Simulated Thermal Power - High	S, D[h]	Q	W[d][e], SA, R[i]	1
c. Neutron Flux - High	S	Q	W[d], SA	1
d. Inoperative	NA	Q	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	S	Q	R[g]	1, 2[j]
4. Reactor Vessel Water Level - Low, Level 3	S	Q	R[g]	1, 2
5. Reactor Vessel Water Level - High, Level 8	S	Q	R[g]	1
6. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
7. Main Steam Line Radiation - High	S	Q	R	1, 2[j]
8. Drywell Pressure - High	S	Q	R[g]	1, 2[k]

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Scram Discharge Volume Water Level - High				
a. Transmitter/Trip Unit	S	Q	R(g)	1, 2, 5(l)
b. Float Switch	NA	Q	R	1, 2, 5(l)
10. Turbine Stop Valve - Closure	S	Q	R(g)	1
11. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	S	Q	R(g)	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	W	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decade during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decade during each controlled shutdown, if not performed within the previous 7 days.

(c) [DELETED]

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 MWD/T using the TIP system.

(g) Calibrate trip unit at least once per 92 days.

(h) Verify measured drive flow to be less than or equal to established drive flow at the existing flow control valve position.

(i) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

(j) Not applicable when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(k) Not applicable when DRYWELL INTEGRITY is not required.

(l) Applicable with any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.
b. Preserve the integrity of the reactor coolant system.
c. Minimize the energy which must be absorbed following a loss-of-coolant accident, and
d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," as approved by the NRC and documented in the SER (letter T. A. Pickens from A. Thadani dated July 15, 1987). The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

ISOLATION ACTUATION INSTRUMENTATION (continued)

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 10 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 10 second delay. It follows that checking the valve speeds and the 10 second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram recirculation pump trip (ATWS-RPT) system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event has been evaluated in General Electric Company report NEDC-32408 dated March 1987. The results of the analysis show that the Grand Gulf ATWS-RPT design provides adequate protection for these events in which the normal scram paths fail.

The ATWS-RPT provides fully redundant trip of the recirculation pump motors so that the pumps coast down to zero speed. This trip function reduces core flow creating steam voids in the core, thereby decreasing power generation and limiting any power or pressure excursions. The Grand Gulf ATWS-RPT design provides compliance with the requirements of the NRC ATWS Rule 10CFR50.62.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 67 TO FACILITY OPERATING LICENSE NO. NPF-29

SYSTEM ENERGY RESOURCES, INC.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated June 30, 1988, as supplemented by letter dated February 19, 1990, System Energy Resources, Inc. (SERI or the licensee), requested an amendment to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1 (GGNS-1). The proposed amendment would change the Technical Specifications (TS) by increasing the surveillance test intervals (STIs) and allowable out-of-service times (AOTs) for the reactor protection system (RPS). These changes are based upon the BWR Owners Group (BWROG) Topical Report NEDC-30851P, "Technical Specification Improvement Analysis for BWR Reactor Protection System," dated May 31, 1985, which provided a safety analysis for increased surveillance test intervals for RPS instrumentation on a generic basis. The NRC staff reviewed NEDC-30851P and issued a Safety Evaluation (SE) on July 15, 1987 approving the report and providing model TS changes. The staff's SE was incorporated into the approved version of the topical report, NEDC-30851P-A, issued April 4, 1988.

The staff's generic SE stated that plant specific application of the generic results would require comparison of the plant specific design with the generic design to show that NEDC-30851P-A is applicable. The licensee's submittal, dated June 30, 1988, included the General Electric Company (GE) Report MDE-80-0485, dated April 1985, which compared GGNS-1 RPS design with that used in NEDC-30851-P. The submittal also provided SERI's response to the plant specific conditions required to be met by the staff's generic SE.

The supplemental information submitted by the licensee's letter dated February 19, 1990, provided supplemental data regarding the drift of RPS instrumentation. The proposed TS were not changed. This submittal did not alter the action previously noticed or affect the initial determination published in the Federal Register on August 1, 1988 (53 FR 28927).

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2.0 EVALUATION

The NRC staff has reviewed the licensee's June 30, 1988 and February 19, 1990 submittals. The proposed TS changes reflect those standard TS revisions contained in NEDC-30851P-A which, based upon probabilistic analyses, justify the identified time extensions by reducing the potential for: (1) unnecessary plant scrams; (2) excessive equipment test cycles; and (3) diversion of personnel and resources to conduct unnecessary testing.

The licensee has extended the generic analysis completed by the BWR Owners Group to GGNS-1 by having General Electric Company complete the required plant specific analysis. As stated in the NRC's SE for Licensing Topical Report NEDC-30851P-A, three conditions must be addressed to justify the applicability of the generic analysis to individual plants when specific facility TS are considered for revision.

1. Confirm the applicability of the generic analysis to the specific facility.

   Licensing Topical Report NEDC-30851P-A, Appendix L, identifies the licensee as a participating utility in the development of the RPS TS improvement analysis. Section 7.4 specifies that although "the evaluation found various differences between the RPS configuration of various plants and the generic plant....the generic results can be applied to plants in the BWROG Technical Specifications Improvement Program." Therefore, the generic analysis contained within the referenced report is applicable to GGNS-1.

2. Demonstrate that the drift characteristics for RPS channel instrumentation are bounded by the assumptions used in NEDC-30851P-A when the functional test interval is extended from monthly to quarterly.

   The additional time interval between tests resulting from the changes described in NEDC-30851P-A, and requested in this submittal, is already factored into the instrument setpoint calculations for the affected instruments. As stated in the Bases to GGNS-1 TS 2.2.1, the difference between each RPS instrument trip setpoint and the allowable value is equal to or greater than the drift allowance assumed for each trip in the plant safety analyses. The setpoint calculations conservatively assume an eighteen month calibration interval and the drift based upon vendor supplied values associated with that interval with no credit taken for the currently specified 31 day functional test. This assumption in the setpoint calculations, therefore, bounds any drift which could be expected over the 92 day functional test interval proposed. By letter dated February 19, 1990, the licensee provided data from surveillance tests at GGNS-1 for a representative sample of RPS instrumentation and demonstrated that instrumentation drift characteristics are bounded for the proposed surveillance test interval. Accordingly, revised instrument setpoints or allowable values are not required to accommodate the longer test intervals requested.

3. Confirm that the differences between the parts of the RPS that perform the trip functions in the plant and those of the base case plant were included in the specific analysis done using the procedures of Appendix K to NEDC-30851P-A.

   In the GE report NEDC-30851P-A, "Technical Specification Improvement Analysis for the Reactor Protection System for GGNS-1," dated April 1985, the generic study completed in Licensing Topical Report NEDC-30851P-A for modifying the RPS was extended to GGNS-1. The GE report utilizes the procedures of Licensing Topical Report NEDC-30851P-A, Appendix K, to identify and evaluate the differences between the parts of RPS that perform the trip functions at GGNS-1 and those of the base case plant. The results indicate that while the RPS configuration for GGNS-1 has several differences compared to the configuration in the base case, the differences and their impact do not significantly affect the applicability of the TS changes developed by the generic efforts of Licensing Topical Report NEDC-30851P-A.

   SERI has reviewed the plant specific report for GGNS (NEDC-30851P-A) and has verified that the differences between the GGNS-1 RPS at the time the analysis was made and the generic RPS were included in the plant specific analysis. Since the plant specific analysis was done (April 1985), the only RPS changes which have occurred make GGNS-1 RPS closer to the generic plant. Therefore, the conclusions reached in NEDC-30851P-A apply to GGNS-1 and the plant-specific changes contained in this request are bounded by both the generic analysis and the NRC's generic safety evaluation.

Based on the staff's evaluation of the licensee's submittal, the staff finds that GGNS-1 has met the plant specific conditions needed to apply the results of GE's Topical Report NEDC-30851P-A to the Grand Gulf Nuclear Station, Unit 1.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on March 16, 1989 (54 FR 11092). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration, which was published in the Federal Register on August 1, 1988 (53 FR 28927), and consulted with the State of Mississippi. No public comments or request for hearing with the were received, and the State of Mississippi had no comments.

The staff has 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 this amendment will not be inimical to the common defense and security, or to the health and safety of the public.

Principal Contributor: L. L. Kintner

Dated: April 24, 1990