June 23, 1989

Docket No. 50-458

Gulf States Utilities
ATTN: Mr. James C. Deddens
Senior Vice President (RBNG)
Post Office Box 220
St. Francisville, LA 70775

Dear Mr. Deddens:

SUBJECT: RIVER BEND STATION, UNIT 1 - AMENDMENT NO.37 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NO. 69105)

The Nuclear Regulatory Commission has issued the enclosed Amendment No.37 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 5, 1988 as supplemented November 30, 1988, January 17 and February 28, 1989.

The amendment modified License Condition 2.C(13) and TS Table 3.3.6-2, Item 1.b, High Power Setpoint, to allow continued operation of the facility with up to 100°F reduction from the rated feedwater temperature of 420°F during the normal fuel cycle.

A copy of our Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Walter A. Paulson, Project Manager
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 37 to License No. NPF-47
2. Safety Evaluation

cc w/enclosures:
See next page

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DOCUMENT NAME: RIVER BEND AMENDMENT

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

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See next page

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River Bend Nuclear Plant

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

GULF STATES UTILITIES COMPANY

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 37
License No. NPF-47

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

   A. The application for amendment by Gulf States Utilities Company (the licensee) dated August 5, 1988, as supplemented November 30, 1988, and January 17 and February 28, 1989, 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 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 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 License Condition 2.C(13) of Facility Operating License No. NPF-47. License Condition 2.C(13) is hereby amended to read as follows:

   (13) Partial Feedwater Heating (Section 15.1, SER)

   The facility shall not be operated with partial feedwater heating beyond the end of the normal fuel cycle without prior written approval of the staff. During the normal fuel cycle, the facility shall not be operated with a feedwater heating capacity which would result in a rated thermal power feedwater temperature less 320°F without prior written approval of the staff.

3. Accordingly, the license is also 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-47 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 37 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. GSU shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

4. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kenneth L. Heitner
for
Frederick J. Hebdon, Director
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 23, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 37

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following page of the Appendix "A" Technical Specifications with the enclosed page. The revised page is identified by Amendment number and contains a vertical line indicating the area of change. The overleaf page is provided to maintain document completeness.

REMOVE PAGE	INSERT PAGE
3/4 3-62	3/4 3-62

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RPCS inoperable and take the ACTION required by Specification 3.1.4.2.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.#

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.#

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.#

NOTES

* With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** OPERABLE channels must be associated with SRM required OPERABLE per Specification 3.9.2.

\# The provisions of Specification 3.0.4 are not applicable.

(a) This function shall be automatically bypassed if detector count rate is $\geq$ 100 cps or the IRM channels are on range 3 or higher.

(b) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(c) This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

(d) This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2
CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD PATTERN CONTROL SYSTEM		
a. Low Power Setpoint	27.5 ± 3% of RATED THERMAL POWER	27.5 ± 7.5% of RATED THERMAL POWER
b. High Power Setpoint	≤ 67.9% of RATED THERMAL POWER	≤ 68.2% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux Upscale		
1) Two Recirculation Loop Operation	≤ 0.66W + 42%*	≤ 0.66W + 45%*
2) Single Recirculation Loop Operation	≤ 0.66W + 36.7%*	≤ 0.66W + 39.7%*
b. Inoperative	NA	NA
c. Downscale	≥5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	< 1×10^5 cps	< 1.6×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 0.7 cps	≥ 0.5 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 division of full scale	≤ 110/125 division of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 division of full scale	≥ 3/125 division of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High - LISN602A	≤ 18.00"	≤ 21.12"
LISN602B	≤ 18.00"	≤ 21.60"
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108% of rated flow	≤ 111% of rated flow

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

**Provided signal to noise ratio is ≥ 2, otherwise setpoint of 3 cps and allowable 1.8 cps.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 37 TO FACILITY OPERATING LICENSE NO. NPF-47

GULF STATES UTILITIES COMPANY

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By letter dated August 5, 1988 as supplemented November 30, 1988, January 17, 1989, and February 28, 1989, Gulf States Utilities Company (GSU) (the licensee) requested an amendment to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The proposed amendment would modify License Condition 2.C(13) and TS Table 3.3.6-2, Item 1.b to allow continued operation of the facility with up to 100°F reduction from the rated feedwater temperature of 420°F during the normal fuel cycle. Planned operation in this mode for the purpose of extending the normal fuel cycle would continue to be prohibited. During the review of the River Bend Station application for an operating license, the NRC staff requested quantitative analysis for operation with partial feedwater heating prior to operation in this mode. GSU committed to provide this analysis prior to this mode of operation.

The licensee's August 5, 1988 application included an analysis performed by the General Electric Company (GE) for operation with feedwater heaters out of service (NEDO-31583, May 1988). This analysis evaluated the operation of River Bend Station with feedwater temperature ranging from 420°F to 320°F. The following items that were considered to be potentially affected by feedwater heater out of service (FWHOS) operation were evaluated by GE: transient response, reactor core thermal-hydraulic stability margins, ECCS thermal-hydraulic analysis, acoustic loads during postulated LOCA events, annulus pressurization loads during postulated LOCA events, containment response and loads during postulated LOCA events, and fatigue usage of feedwater nozzles, spargers, and piping. In addition, GSU provided an analysis of the high power and low power setpoints of the rod control and information system and an evaluation of FWHOS operation with only one recirculation loop operating.

The evaluation of transient response, reactor core thermal-hydraulic stability margins, and ECCS thermal hydraulic analysis may be cycle and fuel-type dependent; thus, these items need to be considered in conjunction with reload analyses. The licensee's August 5, 1988 submittal provided an analysis for the topics for Cycle 2 operation. The licensee's November 18, 1988 reload application for Cycle 3 operation stated that the FWHOS analysis is also consistent with the reload analysis for Cycle 3.

The licensee's November 30, 1988, January 17, 1989 and February 28, 1989 submittals provided clarifying information and did not change the finding of the initial notice, published October 5, 1988 (53 FR 39170) or the scope of the amendment request.

2.0 EVALUATION

2.1 Anticipated Operational Occurrences

All core-wide transients in Chapter 15 of the River Bend Station (RBS) Updated Safety Analysis Report were examined for FWHOS operation. Operation with FWHOS results in decreased feedwater temperature and increased subcooling in the core downcomer region and at the core inlet. The most limiting abnormal operating transients were reevaluated in detail. They are:

1. Generator Load Rejection with Bypass Failure (LRBPF)
2. Feedwater Flow Controller Failure, Maximum Demand (FWCF)
3. Loss of 100°F Feedwater Heating (LFWH)

The reevaluations for the LRBPF and FWCF events were performed at 100% power/100% core flow condition with a rated feedwater temperature of 320°F for Cycle 2. The analysis is also consistent with Cycle 3 operation as stated in the licensee's November 18, 1988 Cycle 3 reload application. The GEMINI/ODYN transient analysis methodology (approved by the staff) described in Amendment No. 11 and supplement to NEDE-24011, "GE Generic Licensing Reload Report", was used to simulate the transient events 1 and 2, above. The delta critical power ratios (CPR) are bounded by the Technical Specification limits with respect to LRBPF and FWCF results.

The 100°F loss of feedwater heating transient was evaluated at 102% power and 100% core flow for Cycle 2. Again, the licensee's August 5, 1988 submittal states that this FWHOS analysis is consistent with the reload analysis for Cycle 3. The resulting CPR for the 100°F loss of Feedwater heating initiated from 320°F is bounded by the 420°F normal feedwater CPR. The General Electric Company "Three-Dimensional BWR Core Simulator," referenced in NEDO-20953-A and approved by the NRC was used to perform the analysis.

2.2 Thermal-Hydraulic Stability

GSU responded to Bulletin 88-07, "Power Oscillations in Boiling Water Reactors (BWRs)" in a letter dated September 8, 1988. The letter addressed the adequacy of the instrumentation, procedures and training at River Bend with regard to stability. Interim guidance on stability actions was issued by the Boiling Water Reactor Owners Group (BWROG) on November 4, 1988. The licensee's December 20, 1988 letter stated that GSU implemented the following actions in plant operating procedures: (1) Scram the reactor manually if core flow is less than 40% and rod line is greater than 100%; (2) Take immediate action if core flow is less than

40% and rod line is between 80% and 100%; (3) Avoid core flow between 40% and 45% and rod line greater than 80%, if possible; and (4) Scram the reactor manually if oscillations occur at core flow less than 40% and rod line above 80%.

A rod line describes the variation of power with recirculation flow for a fixed control rod configuration.

On December 30, 1988, NRC Bulletin No. 88-07, Supplement 1 was issued. This supplement requested the following actions for operating reactors:

(1) Within 30 days of receipt of this supplement, all BWR licensees should implement the GE interim stability recommendations described in Attachment 1 (to the supplement). However, for those plants that do not have effective automatic scram protection in the event of regional oscillations, a manual scram should be initiated under all operating conditions when two recirculation pumps trip (or "no pumps operating") with the reactor in the RUN mode.

(2) The boundaries of Regions A, B, and C shown in Figure 1 of the GE recommendations (Attachment 1 to the supplement) were derived for those BWRs using NRC approved GE fuel. For BWRs using fuel supplied by other vendors, these regions should be adopted in principle, but the power/flow boundaries should be based on existing boundaries that have been previously approved by the NRC. For proposed new fuel designs, the stability boundaries should be reevaluated and justified based on any applicable operating experience, calculated changes in core decay ratio using NRC approved methodology, and/or core decay ratio measurements. There should be a high degree of assurance that instabilities will not occur under any circumstances of operation in Region C.

(3) The BWROG recommendations of Attachment 1 (to the supplement) are ambiguous with respect to permissible conditions for entry of Regions B and C. Although the recommendations state that intentional operation in Region B is not permitted and operation in Region C is permitted only for purposes of fuel conditioning during rod withdrawal startup operations, intentional entry into Region B or C is also allowable in situations where rod insertion or a flow increase is required by procedures to exit Regions A and B after unintentional entry. Licensees should ensure that the procedures and training employed for implementation of these recommendations avoid any similar ambiguity which could lead to operator confusion.

    By letter dated March 3, 1989, the licensee stated that all the action items have been completed and implemented at RBS. The NRC staff finds this acceptable until long-term resolution of the stability issue.

    As stated in Bulletin 88-07, Supplement 1, the NRC staff is working with the BWR Owners Group to develop a generic approach to long-term corrective actions. The staff expects to issue another generic communication that will provide guidance for long-term resolution of the stability issue.

2.3 Loss of Coolant Accident and Related Analyses

2.3.1 ECCS Thermal-Hydraulic Performance

A Loss of Coolant Analysis (LOCA) was performed for RBS for FWHOS operation. Reduction of feedwater temperature results in increased subcooling the vessel thus increasing the mass flow rate out of a LOCA break. However, an increase in initial total system mass and a delay in lower plenum flashing also occur. They act together to decrease the impact of increased flow out of the recirculation line break. As a result of this offsetting effect, the peak cladding temperature (PCT) was shown to be lower than the 2144°F value reported for RBS and below the 2200°F 10 CFR 50.46 cladding temperature limit.

2.3.2 Acoustic and Flow-Induced Loads on Reactor Vessel Internals

In responding to the staff's October 3, 1988 request for additional information regarding the effects of FWHOS on the acoustic and flow-induced loads on reactor internals, the licensee's November 30, 1988 letter provided clarification that both acoustic and flow-induced loads are larger with increased reactor downcomer subcooling. As such, it is expected that these loads will be larger in FWHOS operation than they are in normal operation. A bounding analysis of acoustic and flow-induced loads with FWHOS operation has been performed to confirm that, on a generic basis, the BWR/6 reactor internals are able to withstand the higher loads with sufficient margin. Since the generic BWR/6 analyzed downcomer subcooling values are significantly larger than the maximum RBS value (by about 25 BTU/lbm), sufficient design margins exist for the RBS reactor internals relative to acoustic and flow-induced loads in the FWHOS operation.

2.3.3 Annulus Pressurization Loads

An analysis of the impact of FWHOS operation on the annulus pressurization (AP) loads was performed for River Bend Station. An evaluation of the feedwater line break flow was performed because this break results in the greatest forces upon the reactor pressure vessel and the greatest differentials across the biological shield wall. The break flow for the feedwater line break with FWHOS operation was determined to be less than that presented in the Updated Safety Evaluation Report (USAR) during the inventory depletion period of the feedwater line when the peak AP loads occur. Thus, the loads expected to occur for this event during FWHOS operation are bounded by the normal operation AP loads as calculated in the USAR.

2.3.4 Containment Response

The impact of FWHOS on the containment LOCA response was evaluated for both main steamline and recirculation line breaks over the power/flow range for FWHOS operation. The peak drywell and wetwell pressure and temperature, pool swell, condensation oscillation and chugging loads during FWHOS operation were evaluated.

The peak drywell-to-wetwell differential pressure during FWHOS operation was calculated to occur for a recirculation line break at the maximum vessel subcooling condition on the power/flow map. The licensee's November 30, 1988 submittal stated that the Moody slip flow correlation (NEDO-20533) was used in the analysis of the recirculation line break. The peak differential pressure increased by 0.2 psi compared to the main steamline break design basis accident; however, the resulting differential pressure (18.8 psid) is still below the design differential pressure of 25 psid.

The pool swell, condensation oscillation, and chugging loads evaluated at the worst power/flow condition during FWHOS operation vary slightly over the peak values as presented in Section 6 of the USAR. The analysis concluded that this variation is not significant and that adequate design margins exist with regard to these loads.

The staff finds that the results of the licensee's analysis are acceptable.

2.4 Feedwater Nozzle, Sparger, and Piping Fatigue Usage

The licensee performed an evaluation of the integrity of the feedwater nozzle in RBS for FWHOS operation. Assuming a full, single 18-month cycle operation with FWHOS based on an 80% capacity factor would result in 438 full power days of operation per cycle. This will result in an additional 0.0214 fatigue usage factor over 40 years of continuous FWHOS operation. Thus, the fatigue usage factor will still be well below the limit of 1.0.

An evaluation was also performed by the licensee to examine the impact of FWHOS operation on the feedwater sparger for RBS. Two cases were analyzed to determine the number of days allowable per year (for 40 years) for FWHOS operation without exceeding the fatigue usage limit of 1.0. The results showed that over the 40-year period, the average number of days allowable during an operating year for FWHOS operation decreases with lower feedwater temperature; 256 days and 61 days for rated feedwater temperatures of 370°F and 320°F, respectively. As indicated in the August 5, 1988 submittal the licensee has established a stringent administrative control to track the number of days that the unit is operating with partial feedwater heating and the magnitude of the temperature drop in order to ensure that this limit is not exceeded. According to the administrative control procedure, the operator is required to take daily readings of the feedwater temperature. Any time the temperature drops more than 3% from its normal rated value of 420°F (408°F), a condition report will be written and an evaluation will be performed to calculate the corresponding average number of days allowable during an operating year. This is acceptable to the staff.

In the January 17, 1989 submittal, the licensee has also provided additional information regarding the loading used in the stress analysis and fatigue usage calculation for the feedwater system piping. The previously analyzed load case for the standard BWR/6 plant design bounded FWHOS operation both in temperature and number of cycles expected to occur over the life of the plant. Additional analyses were performed by the licensee and Stone and Webster using existing as-built information for the RBS feedwater piping supports. The existing analysis already considered seventy (70) loss of feedwater heater transients from 425°F to 352°F to occur over the life of the plant. Supplemental analyses were performed which modified the existing transients to consider FWHOS operation from 425°F to 320°F. The supplemental analysis considered the requirements in ASME III, Section NB-3600, and other requirements such as line break evaluation, support capacity, and equipment interface loading. The revised cumulative fatigue usage from the modified transient for piping in areas outside of the break exclusion area is found to be 0.6512. This is less than the allowable limit of 1.0. The revised cumulative fatigue usage for piping within the break exclusion area is 0.0938 which is also less than its allowable limit of 0.10. These results are acceptable.

2.5 High Power Setpoint of the Rod Control and Information System

During FWHOS operation, less steam is generated at the same thermal power and therefore Turbine First Stage Pressure (TFSP) is reduced. This TFSP is utilized for a variety of functions as an indication of core thermal power and provides information to the Rod Control and Information System (RCIS) to initiate control rod block at the high power and low power setpoints. The high power control rod block setpoint (allowable value) is based on the analytical limit of 70% of rated thermal power assumed in the rod withdrawal error analysis (USAR Section 15.4.2.3.3). The high power setpoint provides an input to the rod withdrawal limit to initiate more restrictive control rod movements constraints (1 foot vs. 2 foot withdrawal) at reactor power greater than 70% of rated thermal power.

Even though the Technical Specifications indicate the high power setpoint in percentage of rated thermal power, the turbine first stage pressure which corresponds to the thermal power is used to set the actual high power trip setpoint. The methodology used to determine the high power trip setpoint has been changed to account for various uncertainties in pressure measurement, errors in heat balance calculations and modeling and includes FWHOS effects.

The high power trip setpoint presently in the Technical Specification of 62.5± 3% of rated power has been changed to 67.9% of rated thermal power. The high power setpoint allowable value is changed from 62.5± 7.5% of rated thermal power to 68.2% of rated thermal power. There is no change in the low power setpoint due to change in feedwater temperature. The proposed Technical Specification allowable value and trip setpoint values for the high power control rod block setpoint were calculated by subtracting uncertainty components from the analytical limit of 70% of rated thermal power (corresponds to 461.2 psig TFSP). The allowable value was calculated by subtracting uncertainties due to measurement inaccuracies.

These include process measurement, instrument loop, and calibration inaccuracies. The nominal trip setpoint was calculated by subtracting the uncertainty due to instrument channel drift from the allowable value. The actual in-plant setpoint, which is based on first stage turbine inlet pressure, was then determined by subtracting the FWHOS effect on TFSP (39 psia) from the allowable value.

The proposed values ensure that the analytical limit of 70% rated thermal power will not be exceeded by including the appropriate uncertainties. The proposed change is consistent with the rod withdrawal analysis, which assumed control rod withdrawals would be limited to 1 foot above 70% of rated power. The proposed change is also consistent with the upper allowable value currently specified in the Technical Specification. The proposed Technical Specifications change in Table 3.3.6-2, Item 1.b is acceptable.

2.6 Combined FWHOS and Single Recirculation Loop Operation (SLO)

The licensee evaluated plant operation for combined FWHOS and SLO conditions. The evaluation included rod withdrawal error, Loss of Coolant Accident, Thermal Hydraulic Stability etc. The minimum operating limit CPR for SLO conditions is 1.42. The CPR for limiting pressurization transient (feedwater controller failure) during SLO conditions (0.12) with the additional effect of FWHOS (0.02) results in a total CPR of 0.14. This will result in a minimum operating CPR of 1.28. Therefore, the SLO safety limit of MCPR of 1.08 is still maintained.

The combined operation evaluation concluded that the restrictions imposed for SLO also bound operation in the combined FWHOS and SLO conditions; hence, it is acceptable.

2.7 Summary

Based on the review of the licensee's submittals, the NRC staff concludes that operation with feedwater heaters out of service is acceptable for feedwater inlet temperature down to 320°F during the normal fuel cycle. The NRC staff also finds that the proposed modifications to License Condition 2.C(13) and TS Table 3.3.6-2, Item 1.b are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves 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 exposures. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 23, 1989

Principal Contributors: W. Paulson
A. Lee
G. Thomas