May 31, 1991

Docket No. 50-458

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

Dear Mr. Deddens:

SUBJECT: RIVER BEND STATION, UNIT 1 - AMENDMENT NO. 57 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NO. 76930)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 57 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 June 4, 1990 and supplemented by letters dated February 13, 1991, April 11, 1991, and May 2, 1991.

The amendment revised TSs 4.0.5, 3.4.3.1, and 3.4.3.2 regarding inservice inspection, leak detection, and actions to be taken when leakage limits are exceeded and revised the bases of TS 3/4.4.3.1. The TS changes are based on guidance provided in Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping." Additionally, two editorial changes were made which deleted references to the first refueling outage, which has already been completed.

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

Sincerely,

Original Signed By

Claudia M. Abbate, Project Manager
Project Directorate IV-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 57 to NPF-47
2. Safety Evaluation

cc w/enclosures:
See next page
SEE PREVIOUS CONCURRENCE

CP-1

OFC	:PDIV-2/LA *	:PDIV-2/PM *	:EMCB:BC *	:OGC	:PDIV-2/(A)D	:	:
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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. 57
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 June 4, 1990 and supplemented by letters dated February 13, 1991, April 11, 1991, and May 2, 1991, 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 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) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 57 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.

3. The license amendment is effective as of date of issuance to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick Jr

George F. Dick, Acting Director
Project Directorate IV-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: May 31, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 57

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

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

REMOVE	INSERT
3/4 0-3	3/4 0-3
3/4 4-10	3/4 4-10
3/4 4-11	3/4 4-11
3/4 4-12	3/4 4-12
B 3/4 4-3	B 3/4 4-3
	B 3/4 4-3a
B 3/4 4-4	B 3/4 4-4

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection Program (ISI) for piping susceptible to Intergranular Stress Corrosion Cracking (IGSCC) shall be performed in accordance with the NRC positions included in Generic Letter 88-01.

REACTOR COOLANT SYSTEM

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings:

	Low-Low Set Function Setpoint* (psig) ± 15 psi	
Valve No.	Open	Close
1B21*RVF051D	1033	926
1B21*RVF051C	1073	936
1B21*RVF051B	1113	946
1B21*RVF051G	1113	946
1B21*RVF047F	1113	946

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable relief valve function and the low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the low-low set function of more than one of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

c. With either low-low set function pressure actuation trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within 7 days; otherwise, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2.1 The low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit setpoint, at least once per 31 days.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

REACTOR COOLANT SYSTEM

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The drywell atmosphere particulate radioactivity monitoring system,

b. The drywell and pedestal floor sump drain flow monitoring systems,

c. Either the drywell air coolers condensate flow rate monitoring system or the drywell atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With leak detection systems 'a' and/or 'c' inoperable operation may continue for up to 30 days provided grab samples are obtained and analyzed at least once per 24 hours for the inoperable radiation monitors; otherwise, be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the drywell floor and/or pedestal sump drain flow monitoring subsystem inoperable, operation may continue for up to 24 hours otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Drywell atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. The sump drain flow monitoring systems-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

c. Drywell air coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

d. Flow testing the drywell floor drain sump inlet piping for blockage at least once per 18 months.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage (averaged over any 24-hour period).

d. 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm leakage at a reactor coolant system pressure of 1025 ± 15 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

e. 2 gpm UNIDENTIFIED LEAKAGE increase within any period of 24 hours or less (Applicable in OPERATIONAL CONDITION 1 only).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two other closed manual, deactivated automatic or check* valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm point at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours. The provisions of Specification 3.0.4 are not applicable.

e. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than the limits in e., above, within 4 hours identify the source of leakage as not IGSCC susceptible material or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

* Which have been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric particulate radioactivity at least once per 12 hours,

b. Monitoring the sump flow rates at least once per 12 hours,

c. Monitoring the drywell air coolers condensate flow rate at least once per 12 hours, and

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 including paragraph IWV-3427(B) of the ASME Code and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

SAFETY/RELIEF VALVES (Continued)

Demonstration of the safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of 5 valves and lowering the opening setpoint of 2 valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973. In conformance with Regulatory Guide 1.45, the atmospheric gaseous radioactivity system will have a sensitivity of 10^{-6} µCi/cc.

The drywell and pedestal floor sump drain flow monitoring systems consist of 2 sumps with one level transmitter and two 100% pumps each. The level transmitters feed Main Control Room level indicators as well as various automatic control systems. Each of the automatic systems calculate leakage, control pumps and provide annunciation. The leak rate may be determined by the automatic system or a manual procedure through the use of the level indication and pump control switches located in the main control room. The substitution of grab samples for the drywell particulate and gaseous monitors is to allow for continued monitoring of the function while normal components are inoperable.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage, due to equipment design and the detection capability of the instrumentation for determining system leakage, was also considered. The evidence obtained from experiments suggests that, for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.3.2 OPERATIONAL LEAKAGE (Continued)

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low; thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

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

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, 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 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomena which may occur following changes in THERMAL POWER.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 57 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 June 4, 1990, and supplemented by letters dated February 13, 1991, April 11, 1991, and May 2, 1991, Gulf States Utilities Company (GSU) (the licensee) requested an amendment to Facility Operating License No. NPF-47 for the River Bend Station (RBS), Unit 1. The proposed amendment would revise Technical Specifications (TSs) 4.0.5, 3.4.3.1, and 3.4.3.2 in accordance with the guidance provided in Generic Letter (GL) 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping." GL 88-01 required licensees to: (1) provide plans regarding pipe replacement or other measures taken to mitigate IGSCC and provide long-term reliability, (2) implement an inservice inspection (ISI) program for austenitic stainless steel piping which conforms to the positions in GL 88-01, (3) implement a TS change to include the ISI of austenitic stainless steel piping in accordance with GL 88-01, (4) implement TS changes related to leakage detection in conformance with GL 88-01, and (5) notify the NRC when flaws are identified that do not meet IWB-3500 criteria of Section XI of the American Society of Mechanical Engineers (ASME) code or when a change in the condition of a cracked weld occurs. GSU responded by letters dated July 21, 1988 and May 12, 1989. By letter dated January 25, 1990, the NRC forwarded its Safety Evaluation of the GSU response and requested TS changes be submitted that meet the staff's positions in GL 88-01. The June 4, 1990, February 13, 1991, April 11, 1991, and May 2, 1991, letters provided those TS changes. The April 11, 1991, and May 2, 1991, letters provided information that did not change the initial proposed no significant hazards consideration determination. Specifically, the proposed changes are:

(1) TS 4.0.5 would be modified to state that ISI for piping susceptible to IGSCC would be performed in accordance with GL 88-01;

(2) TS 3.4.3.1 would be modified to reflect actions to be taken when leak detection systems are inoperable;

(3) TS 3.4.3.2 would be modified to reflect a new unidentified leakage rate limit and actions to be taken when that limit is exceeded; and

(4) the BASES of TS 3/4.4.3.1 would be revised to reflect the TS changes.

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Two editorial changes were also proposed which would delete references to the first refueling outage, which has already been completed.

2.0 EVALUATION

The proposed change to TS 4.0.5, "Applicability, Surveillance Requirements," is based on the staff position in GL 88-01 on inspection methods and personnel and would add a requirement that ISI be performed in accordance with the NRC staff positions in GL 88-01. GSU also committed to revise the Updated Safety Analysis Report (USAR) and the RBS procedures which involve inservice inspection of piping welds. The proposed change is considered an enhancement to the current ISI program.

The proposed change to the ACTION statement of TS 3.4.3.1, "Leakage Detection Systems," is based on the staff position in GL 88-01 on leakage detection and would separate the action to be taken when leakage detection systems are inoperable into two parts. The first part of the ACTION statement would allow operation up to 30 days with the drywell atmosphere particulate radioactivity monitoring system, or either the drywell air coolers condensate flow rate monitoring system or the drywell atmosphere gaseous radioactivity monitoring system inoperable, provided a "grab" sample is taken and analyzed once every 24 hours. Otherwise, the plant would be required to be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. The second part of the proposed ACTION statement would allow continued operation for up to 24 hours with the drywell floor and/or pedestal sump drain flow monitoring systems inoperable. If the systems could not be restored within 24 hours, the plant would be required to be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

The drywell leak detection system is described in Section 5.2.5.1.1, "Detection of Leakage Within the Drywell," of the USAR and consists of airborne and particulate radioactive monitoring systems, the drywell cooler condensate flow monitoring system, and the sump drain flow monitoring system. The variables are continuously monitored and indicated in the control room and comply with Positions C.3 and C.7 of Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." No changes to the TSs regarding the radioactive monitoring systems or drywell cooler condensate flow system are proposed.

The sump drain flow monitoring system is divided into two subsystems, each having a 600 gallon capacity sump. These are: the floor sump, located in the general drywell space at the 81 foot elevation and the pedestal sump, located under the reactor vessel at the 73 foot elevation. Both have a sensitivity for detection of leakage increase of one gallon per minute (gpm) in one hour. This complies with Position C.3 of RG 1.45. A programmable controller (PC) monitors the sumps and alerts the operators in the control room of increased unidentified leakage. This alert occurs before the TS limit of five gpm unidentified leakage is reached and an alarm sounds in the control room. When the PC is inoperable or during a surveillance procedure, the leakage flow rate, which is the sum of both sump leakage flows, is calculated manually by the operators using Standard Operating Procedure (SOP)-0104. GSU is confident that the manual method of calculation yields complete and accurate information that demonstrates compliance with the TSs and does not inhibit or prevent the operators from identifying the leakage rates. The proposed TS changes would allow 24 hours for repair if either of the sump subsystems become inoperable. During that time, the radiation monitoring systems would be capable of detecting a one gpm increase in leakage. GSU has determined the radiation monitors provide adequate information regarding leakage rates for the 24-hour period the sumps are allowed out-of-service. The radiation monitoring systems, discussed in Section 11.5.2.1.3.4, "Containment and Drywell Atmosphere," of the USAR, are seismically qualified, powered from a safety-related Class 1E source, and have an acceptable sensitivity and range. The systems comply with Position C.6 of RG 1.45.

The proposed change to TS 3.4.3.2, "Operational Leakage," is based on the NRC staff position in GL 88-01 on leak detection and would place the plant in a limiting condition for operation when a two gpm increase in unidentified leakage occurs in OPERATIONAL CONDITION 1. GSU determined that during startup, leakage rates could increase as part of the normal filling, flow, and pressurization of the reactor coolant system, and would not be indicative of leakage resulting from IGSCC. Additionally, in any other operational condition the probability of the occurrence of IGSCC would be low due to the short period of time the plant is in other operational modes and the conservative condition of the plant. When the LCO is exceeded, the proposed ACTION statement would require that within 4 hours the source of the increased leakage be identified as not being IGSCC or the plant would be required to be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. The drain sump monitoring system will be used to identify leakage rates. If the PC is inoperable, operators would use the same procedure, SOP-0104, to calculate the unidentified leakage rate, and the total leakage (averaged over 24 hours) and maintain the 24-hour total leakage in order to confirm compliance with TS 3.4.3.2. GSU plans on revising this procedure to reflect the additional ACTION statement. GSU is confident that the manual method of calculation yields complete and accurate information that demonstrates compliance with the TSs and does not inhibit or prevent the operators from identifying the leakage rate. During the time when the drywell and/or pedestal sump drain flow monitoring systems are inoperable and the reactor coolant system leakage limits need to be verified, GSU proposes to use the radiation monitoring systems to identify one gpm leakage rate increases. As discussed above, GSU has determined the radiation monitors provide adequate information regarding increases in leakage rates when other leakage detection monitoring systems are inoperable and comply with the staff positions in RG 1.45.

The staff position in GL 88-01 on monitoring frequency suggests a frequency of 4 hours, however, GSU proposes to continue to monitor reactor system leakage once per 12 hours as currently identified in TS Surveillance Requirement 4.4.3.2.1. GSU bases this proposal on the fact that RBS implements a 12-hour operations shift and any increase in leakage will be alerted in the control room at one gpm based on the radiation monitors. If the sump flow leakage increased to the five gpm setpoint, an alarm would sound and the plant would be required to shutdown. Therefore, there are no proposed changes regarding monitoring frequency and TS 4.4.3.2.1.

The two editorial changes to TS Surveillance Requirement 4.4.3.1.d. and 4.4.3.2.2.a. delete the references and footnotes regarding tests that were not required to be performed or tests that were to be performed during the first refueling outage. The first refueling outage has been completed, therefore, the references and footnotes may be deleted.

Because of the changes to TS 3.4.3.1, GSU proposed changes to the BASES section 3/4.4.3.1. The BASES provide the operators background, additional information and guidance on the TSs.

Based on the review of the GSU amendment request, the guidance provided in GL 88-01 and RG 1.45, and the applicable sections of the USAR, Safety Evaluation Report, and Standard Review Plan, the NRC staff finds the proposed changes acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC 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 exposure. 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 (56 FR 13664). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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.

5.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Claudia M. Abbate, PDIV-2

Date: May 31, 1991