January 29 1996

Mr. John R. McGaha, Jr.
Vice President - Operations
Entergy Operations, Inc.
River Bend Station
P. O. Box 220
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NO. M92482)

Dear Mr. McGaha:

The Commission has issued the enclosed Amendment No. 87 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 May 30, 1995, as supplemented by letters dated November 20 and December 12, 1995.

The amendment revises the TSs for the drywell to permit bypass testing on a 10-year frequency with increased testing if performance degrades, changes the drywell air lock testing and surveillance requirements, deletes action notes for the drywell air lock and drywell isolation valves when the bypass leakage limit is not met, and deletes the specific leakage limits for the drywell air lock seal.

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

Sincerely,

ORIGINAL SIGNED BY
David L. Wigginton, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures: 1. Amendment No. 87 to NPF-47
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

GULF STATES UTILITIES COMPANY**

CAJUN ELECTRIC POWER COOPERATIVE AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. NPF-47

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

A. The application for amendment by Gulf States Utilities* (the licensee) dated May 30, 1995, as supplemented by letters dated November 20 and December 12, 1995, 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

---

* EOI is authorized to act as agent for Gulf States Utilities Company, which has been authorized to act as agent for Cajun Electric Power Cooperative, and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

**Gulf States Utilities Company, which owns a 70 percent undivided interest in River Bend, has merged with a wholly owned subsidiary of Entergy Corporation. Gulf States Utilities Company was the surviving company in the merger.

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) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 87 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

David L. Wigginton, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 29, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 87

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

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

REMOVE	INSERT
3.6-61	3.6-61
3.6-62	3.6-62
3.6-63	3.6-63
3.6-64	3.6-64
3.6-65	3.6-65
3.6-66	3.6-66
3.6-67	3.6-67

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.5.1.3	Verify bypass leakage is less than or equal to the bypass leakage limit. However, during the first unit startup following bypass leakage testing performed in accordance with this SR, the acceptance criterion is ≤ 10% of the drywell bypass leakage limit.	-----NOTE----- SR 3.0.2 is not applicable for extensions > 12 months. -------------- 24 months following 2 consecutive tests with bypass leakage greater than the bypass leakage limit until 2 consecutive tests are less than or equal to the bypass leakage limit AND 48 months following a test leakage bypass leakage greater than the bypass leakage limit AND 120 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.5.1.4	Visually inspect the exposed accessible interior and exterior surfaces of the drywell.	Once prior to performance of each Type A test required by SR 3.6.1.1.1
SR 3.6.5.1.5	Verify seal leakage rate when the gap between the door seals is pressurized to ≥ 3 psid.	Once within 72 hours after each drywell air lock door closing
SR 3.6.5.1.6	Verify drywell air lock leakage by performing an air lock barrel leakage test at ≥ 3 psid.	24 months

3.6 CONTAINMENT SYSTEMS

3.6.5.2 Drywell Air Lock

LCO 3.6.5.2 The drywell air lock shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----------------------------------NOTE-----------------------------------

Entry and exit is permissible to perform repairs of the affected air lock components.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One drywell air lock door inoperable.	------------NOTES------------ 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls. ----------------------------	
	A.1 Verify the OPERABLE door is closed. <u>AND</u>	1 hour
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Lock the OPERABLE door closed. AND A.3 Verify by administrative means the OPERABLE door is locked closed.	24 hours Once per 31 days
B. Drywell air lock interlock mechanism inoperable.	------------NOTES------------ 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible under the control of a dedicated individual. ----------------------------- B.1 Verify an OPERABLE door is closed. AND B.2 Lock an OPERABLE door closed. AND B.3 Verify by administrative means an OPERABLE door is locked closed.	 1 hour 24 hours Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Drywell air lock inoperable for reasons other than Condition A or B.	C.1	Verify a door is closed.	1 hour
	AND		
	C.2	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.2.1	Deleted	
SR 3.6.5.2.2	Verify drywell air lock seal air flask pressure is $\geq$ 75 psig.	7 days
SR 3.6.5.2.3	-----------------NOTE-------------------- Only required to be performed upon entry into drywell. --- Verify only one door in the drywell air lock can be opened at a time.	24 months
SR 3.6.5.2.4	Deleted	
SR 3.6.5.2.5	Verify, from an initial pressure of 75 psig, the drywell air lock seal pneumatic system pressure does not decay at a rate equivalent to > 0.67 psig for a period of 24 hours.	18 months

3.6 CONTAINMENT SYSTEMS

3.6.5.3 Drywell Isolation Valves

LCO 3.6.5.3 Each Drywell Isolation Valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

------------------------------------NOTES------------------------------------

1. Penetration flow paths, except for the 24 inch purge valve penetration flow path, may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by Drywell Isolation Valves.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more penetration flow paths with one drywell isolation valve inoperable.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	8 hours
	<u>AND</u>		(continued)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By application dated May 30, 1995, as supplemented by letters dated November 20 and December 12, 1995, Entergy Operations, Inc. (EOI) (the licensee) requested changes to the Technical Specifications (TSs) (Appendix A to Facility Operating License No. NPF-47) for the River Bend Station, Unit 1. (EOI also submitted a letter dated June 20, 1995, to correct a clerical error in the affirmation accompanying the May 30, 1995 application.) The proposed changes would revise the TSs as follows:

1. changes the drywell bypass test surveillance interval from 18 months to 10 years with an increased testing frequency required if performance degrades;

2. changes the drywell air lock testing to, a) move the leakage rate surveillance to the drywell limiting condition for operation (LCO), b) delete the specific overall leakage limit for the air lock, c) delete the note that an inoperable air lock door does not invalidate the previous air lock leakage test, d) move the Note to the bases that required the air lock leakage test at 3 psid be preceded by pressurizing the air lock to 19.2 psid, and e) change the surveillance frequency for the air lock leakage test and interlock test from 18 months to 24 months;

3. deletes the Actions Notes in the drywell air lock LCO and the drywell isolation valve LCO that identifies that the Actions required by the drywell LCO must be taken when the drywell bypass leakage limit is not met; and

4. deletes the requirement for the drywell air lock seal leakage rate to meet a specific leakage limit.

The November 20 and December 12, 1995, letters supplemented the May 30, 1995, application and changed the original proposal which was to test the drywell every 5 years. All these actions were noticed in the Federal Register on December 6, 1995 (60 FR 62490). The licensee provided additional information by letter dated December 12, 1995, for their qualitative assessment of the drywell to support their assurance of continued operability. This information provided clarification and did not change the initial no significant hazards consideration determination.

2.0 GENERAL BACKGROUND

By letter dated October 22, 1993, (EOI), the licensee for Grand Gulf Nuclear Station, a BWR/6 with a Mark III containment, proposed changes to the Grand Gulf TSs to revise the test interval for drywell bypass leakage rate testing and to revise the surveillance for drywell air lock testing. The licensee also proposed to relocate certain drywell air lock tests from the TSs to administrative control.

The licensee supplemented this submittal by submittals dated February 10 and 14, 1995. These submittals proposed a modification to the October 22, 1993, submittal to permit a one-time postponement of the drywell bypass leakage rate test until entry into MODE 2 on the first plant startup from RFO 8; that is, the test would not be performed during Refueling Outage (RFO) 7. The request for a postponement was based on the good previous performance (low bypass leakage) of the Grand Gulf Nuclear Station drywell.

On February 16, 1995, the NRC issued Amendment No. 119 to the Grand Gulf Nuclear Station operating license which approved a one cycle postponement of the drywell bypass leakage rate test. This postponement was for the purpose of providing more time to complete the review of the October 22, 1993, submittal.

Illinois Power Company, the licensee for the Clinton Power Station, another BWR/6 with a Mark III containment, requested a similar change to the TSs in letters dated August 12 and October 14, 1994. By letter dated March 1, 1995, the staff agreed to a postponement of the drywell bypass leakage test for one cycle to consider the Clinton proposal.

By letter dated May 30, 1995, EOI, as the licensee for River Bend Station, also a BWR/6 with a Mark III containment, proposed changes to the TSs to allow the drywell bypass leakage rate tests to be performed on five year intervals.

Because of the interest of these BWR/6 licensees, the NRC staff requested that the BWR/6 licensees work together on a common proposal.

As a first step toward a common proposal, the staff, on September 12, 1995, met with representatives of EOI and representatives of the licensees of the other BWR/6s to discuss increasing the drywell bypass leakage test interval.

Subsequently, the staff received a revised November 20, 1995, proposal from EOI for the River Bend Station and Grand Gulf Nuclear Station.

The revised submittal proposes an increase in the test interval of the drywell bypass leakage rate test and several changes to the drywell air lock surveillance for Grand Gulf Nuclear Station and River Bend Station. The staff's evaluation of the drywell bypass leakage test proposals is discussed in Section 2.0 of this safety evaluation. An evaluation of the proposed drywell air lock TSs changes are discussed in Section 3.0.

The staff has concluded, for the reasons given in this evaluation, that the test interval for the drywell bypass leakage rate test may be extended from 18 months to 10 years. The staff also finds the proposed changes to drywell air lock TSs to be acceptable.

2.1 Drywell Bypass Leakage Discussion

2.1.1 Description of Drywell Safety Function

The Mark III is a pressure suppression containment which is designed to condense steam and contain fission products released during a loss-of-coolant (LOCA) accident. The Mark III containment is only used in this country with the BWR/6 reactor design. The effectiveness of the pressure suppression containment depends on the ability to condense steam released from the primary system during a LOCA. Condensation of the steam precludes overpressurization of the containment. The steam is condensed by directing its flow through a vent system from the drywell, through the suppression pool, to the containment.

The design of the Mark III containment makes allowance for a given amount of steam to bypass the suppression pool and enter the containment without being condensed by the suppression pool. If the bypass leakage were too large, the containment design pressure could be exceeded. There is some margin above the design pressure before the containment would fail; however, if the amount of steam leaking into the containment were large enough, not only could the containment fail, but bypassing the suppression pool could result in a radiation source term much larger than would otherwise be the case.

2.1.2 Drywell Bypass Limit

Grand Gulf Nuclear Station Updated Final Safety Analysis Report (UFSAR) Section 6.2.1.1.5.2 defines allowable bypass leakage as the amount of steam which could bypass the suppression pool without exceeding the design containment pressure of 15 psig. For River Bend Station, the definition is the same and the containment design pressure is also 15 psig. This allowable bypass leakage is determined by examining a spectrum of LOCA break sizes. The allowable leakage is expressed in terms of the parameter $A/\sqrt{K}$ where

A = Flow area of the leakage path, ft^2
K = Geometric and friction loss coefficient, dimensionless.

This parameter is dependent on the geometry of the drywell leakage paths with only a slight flow dependence, which is negligible.

The Grand Gulf Nuclear Station TSs require that, prior to startup after performing a drywell bypass leakage rate test, the drywell bypass leakage rate shall be ≤ 10% of the bypass leakage limit. The drywell bypass leakage rate limit is given in the TSs Bases as $A/\sqrt{K} = 0.9\ ft^2$. The corresponding number for River Bend is 1.0 ft^2 and is specified in the bases for surveillance requirement 3.6.5.1.3.

The drywell bypass limit is based on a very small reactor system break which will not automatically result in a reactor depressurization. It is assumed that, after the break has occurred, the operator shuts the reactor down at a cooldown rate of 100°F/hr. At this rate, it takes 6 hours to depressurize the reactor and terminate break flow to the drywell. It is assumed in the Grand Gulf analysis that one containment spray loop is initiated. Passive containment heat sinks (listed in UFSAR Table 6.2-9) are also credited. This is an important assumption. Without containment spray and containment heat sinks the allowable $A/\sqrt{K}$ is only 0.048 ft^2. River Bend Station does not have a containment spray system. Instead, safety-related fan coolers are utilized to reduce containment pressure along with the heat sinks listed in Table 6.2-27 of the River Bend Station UFSAR.

The design basis leakage for Grand Gulf Nuclear Station corresponds to approximately 35,000 scfm. The corresponding design basis leakage for River Bend Station is approximately 46,000 scfm (November 20, 1995 submittal). These leakage rates are three orders-of-magnitude greater than the primary containment leakage rate (stated in the November 20, 1995, submittal to be less than 10 scfm for Grand Gulf).

Preoperational drywell bypass leakage rate tests were performed at Grand Gulf Nuclear Station and River Bend Station. These test were performed at drywell design pressure (30 psig for Grand Gulf Nuclear Station and 25 psig for River Bend Station) with the drywell isolated from the containment by capping the horizonal vents. The results of these tests at both plants were acceptable. This is discussed further below.

The Grand Gulf Nuclear Station and River Bend Station TSs currently require that a test be performed at least every 18 months to measure the drywell bypass leakage rate. The test is performed at a pressure difference of 3 psid between the drywell and the wetwell. This pressure difference corresponds to the difference in the head of water when the water level in the vent annulus is depressed to the top of the upper row of vents (see Grand Gulf Nuclear Station UFSAR Section 6.2.1.1.5.4, River Bend Station UFSAR Section 6.2.1.1.3.4). It is also the calculated pressure for the design basis accident for drywell bypass leakage.

The November 20, 1995, submittal proposes to increase this test interval to one test in 10 years for both River Bend Station and Grand Gulf Nuclear Station.

2.1.3 Pertinent Differences Between Grand Gulf and River Bend

Although both Grand Gulf Nuclear Station and River Bend Station are BWR/6 reactors with Mark III containments, and the design of the two reactors is the same in many respects, there are several differences which are important in the context of drywell bypass leakage.

Grand Gulf Nuclear Station has a safety-related containment spray system which can mitigate an increase in containment pressure due to drywell bypass leakage. River Bend Station does not have a containment spray system. It utilizes containment fan coolers to perform the same function.

The tables provide some of the pertinent design information for the two facilities.

GRAND GULF DRYWELL DESIGN PARAMETERS

DRYWELL DESIGN PRESSURE	30 psi
PRIMARY CONTAINMENT DESIGN PRESSURE	15 psi
DESIGN DRYWELL BYPASS LEAKAGE-SMALL BREAK LOCA WITH ONE CONTAINMENT SPRAY	0.9 ft^2 ($A/\sqrt{K}$) approx. 35,0000 scfm at 3 psid
DESIGN DRYWELL BYPASS LEAKAGE-LARGE BREAK LOCA WITH NO CONTAINMENT SPRAY	4.3 ft^2 ($A/\sqrt{K}$) approximately 167,0000 scfm at 3 psid approximately 840,0000 scfm at 30 psid

RIVER BEND DRYWELL DESIGN PARAMETERS

DRYWELL DESIGN PRESSURE	25 psi
PRIMARY CONTAINMENT DESIGN PRESSURE	15 psi
DESIGN DRYWELL BYPASS LEAKAGE-SMALL BREAK LOCA WITH UNIT COOLERS	1.15 ft^2 ($A/\sqrt{K}$) * approx. 46,0000 scfm at 3 psid (design) approx. 40,110 scfm at 3 psid (Technical Specification)
DESIGN DRYWELL BYPASS LEAKAGE-LARGE BREAK LOCA WITH UNIT COOLERS	10.3 ft^2 ($A/\sqrt{K}$) approx. 413,0000 scfm at 3 psid

* UFSAR calculations done at $A/\sqrt{}$ = 1.15 ft^2. The TSs value is 1.0 ft^2.

2.1.4 Licensee Proposal for Extending the Drywell Bypass Leakage Test Interval

TSs Surveillance 3.6.5.1.1 for Grand Gulf Nuclear Station and TS 3.6.5.1.3 for River Bend Station require that drywell bypass leakage be verified to be less than the limit every 18 months. The Bases specify a criterion of $A/\sqrt{K}$ = 0.9 ft^2 for Grand Gulf Nuclear Station and 1.0 ft^2 for River Bend Station.

The licensee proposes to change the surveillance frequency for this test to 10 years for both Grand Gulf Nuclear Station and River Bend Station.

2.2 Evaluation

The staff's acceptance of the proposed 10-year test interval is based on the licensee's capability to assure that the likelihood of significant bypass leakage is acceptably low. This is based on the design of the drywell and its penetrations, the TSs and administrative controls in place for both facilities, the results of previous leakage tests, as well as deterministic and risk calculations. The staff gave considerable weight in its evaluation to the licensee's commitment to assess the drywell leakage at least once per cycle to assure that the drywell remains operable for Grand Gulf Nuclear station and River Bend Station.

2.2.1 Overview

The drywell contains penetrations for piping systems; electrical cables for power, control and instrumentation; and a personnel air lock. Grand Gulf Nuclear Station has a drywell equipment hatch and the River Bend Station drywell has a combination equipment hatch and personnel door assembly (see River Bend Station UFSAR Section 3.8.2.1.3.1). Piping penetrations have automatic or remote manual isolation valves or are required to be in the closed position when drywell integrity is required. The electrical penetrations contain a sealing medium to limit leakage. The TSs specify leakage rate testing of the drywell air lock and specify the leakage rate criteria. The licensee proposes to modify the air lock requirements. An evaluation of the licensee's proposal for revising the drywell air lock TSs is provided in Sections 3.2 and 3.3 of this evaluation.

2.2.2 Operating Experience

The tables below provide a summary of the drywell bypass leakage rate testing experience at Grand Gulf Nuclear Station and River Bend Station. The operating experience has been good. The maximum value of bypass leakage for both plants was at Grand Gulf Nuclear Station in June 1984 when a bypass leakage of 7.43% of the design limit was measured.

A total of sixteen drywell bypass leakage rate tests have been performed at both Grand Gulf Nuclear Station and River Bend Station during a time interval of 13 years (a total of 19 years of commercial operation at both units). Grand Gulf Nuclear Station, prior to commercial operation, in March of 1983, failed a drywell bypass leakage rate test. This was due to a partially open vent valve on a vendor supplied compressor and two open electrical conduits. These conduits were open as a result of ongoing construction activities and were scheduled to be sealed closed. However, this was overlooked before the test was performed. These penetrations were subsequently sealed and the drywell bypass leakage was measured to be within acceptable limits. The staff considers this incident to have little bearing on current operation since it is extremely unlikely that the circumstances could be repeated. The electrical penetrations are now permanently sealed, and even if an electrical penetration were reopened for some reason, the level of attention and procedural controls is much higher with the plant in an operational status as opposed to being under construction.

RESULTS OF DRYWELL BYPASS LEAKAGE TESTS
GRAND GULF NUCLEAR STATION

TEST DATE	LEAKRATE scfm	RATIO OF LEAKAGE RATE TO DESIGN LIMIT %	CALCULATED A/√K ft^2
1/82	611	1.75	0.016
3/83	1621	4.63	0.042
6/84	2599	7.43	0.067
11/85	2315	6.61	0.060
11/86 (RFO 1)	1568	4.48	0.040
12/87 (RFO2)	1500	4.29	0.039
4/89 (RFO 3)	1631	4.66	0.042
11/90 (RFO 4)	1591	4.55	0.041
5/92 (RFO 5)	618	1.77	0.016
11/93 (RFO 6)	869	2.48	0.022

RESULTS OF DRYWELL BYPASS LEAKAGE TESTS
RIVER BEND STATION

TEST DATE	LEAK-RATE scfm	RATIO OF LEAKAGE RATE TO DESIGN LIMIT %	CALCULATED A/√k ft^2
4/85	562	1.2	0.014
12/87	602	1.3	0.015
5/89	10	0.022	0.00025
11/90	345	0.75	0.00861
8/92	754	1.6	0.0188
6/94	421	0.91	0.0105

In addition to reviewing the leakage history of the drywells, the staff reviewed the drywell operating experience at all four domestic BWR/6 facilities to determine if there were any operating issues which would indicate that extending the test interval may not be appropriate. None was identified.

2.2.3 Drywell Structure

Section D of the licensee's November 20, 1995, submittal discusses leakage considerations related to the drywell structure. During preoperational testing the drywell was pressurized in large increments to its design pressure (30 psig for Grand Gulf Nuclear Station and 25 psig for River Bend Station) while deflections and strains and concrete crack patterns in the structure were recorded. The results for Grand Gulf Nuclear Station showed that the structure was not stressed as much as predicted and there were no signs of concrete cracking (Grand Gulf Nuclear Station UFSAR 3.8.3.7). The River Bend Station drywell did experience some cracking of the concrete which the licensee characterized as insignificant. The licensee stated in the November 20, 1995, submittal that visual inspections of the accessible drywell surfaces that have been performed since the preoperational tests have not detected additional cracking or other abnormalities in the drywell structure.

During the drywell bypass leakage rate test, the drywell is pressurized to only 3 psid. Thus, the staff expects no significant challenge to the integrity of the drywell structure. This is verified by a statement in the November 20, 1995, submittal that "[v]isual inspections of the drywell surface that have been performed since the [preoperational] structural tests have not revealed the presence of additional cracking or other abnormalities".

The Grand Gulf Nuclear Station and River Bend Station TSs require that the exposed accessible interior and exterior surfaces of the drywell be inspected prior to the performance of each 10 CFR Part 50, Appendix J Type A test. Grand Gulf Nuclear Station currently has an exemption from Appendix J which, among other things, permits Type A tests to be performed on a 10-year interval under specified conditions. Thus, Grand Gulf Nuclear Station is currently required to perform this inspection on an interval no greater than once per 10 years.

By letter dated December 19, 1995, the licensee was granted an amendment to the River Bend license to permit leakage rate testing of the primary containment according to Option B to 10 CFR Part 50, Appendix J which became effective on October 26, 1995. Option B permits Type A testing to be done, when justified by previous good performance, on a 10-year interval. The River Bend Station TSs also specify that drywell visual inspections be performed prior to each Type A test. Thus, the licensee will also visually inspect the River Bend Station drywell on an interval no greater than once per 10 years.

The staff does not consider leakage through the drywell structure to be a significant concern in extending the drywell bypass leakage rate testing frequency for either Grand Gulf Nuclear Station or River Bend Station and considers the 10-year visual inspection frequency to be adequate.

It is conceivable that the licensee may, at some time, modify the drywell structure or some pressure retaining component of the drywell. The Bases to surveillance requirement 3.0.1 state that

> upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE.

The staff considers this to be sufficient to assure that the drywell remains capable of performing its safety function following maintenance.

2.2.4 Piping Penetrations

Lines which penetrate the drywell contain drywell isolation valves. These valves prevent leakage from the drywell into the primary containment. The isolation valves on those lines which penetrate the primary containment as well as the drywell are included in the category of primary containment isolation valves. Primary containment isolation valves are tested according to the requirements of 10 CFR Part 50 Appendix J. Appendix J defines a total leakage rate limit for the containment isolation valves and other penetrations. There is no corresponding limit for the drywell isolation valves. In fact, the drywell isolation valves are not required to be separately leak tested.

A table of drywell isolation valves for the Grand Gulf Nuclear Station is given in a table included in a December 15, 1995, letter from the licensee. Table 6.2-51 of the River Bend Station UFSAR provides the same information for River Bend Station.

The magnitude of allowable drywell bypass leakage makes it unlikely that it will be exceeded due to leakage through a closed drywell isolation valve or valves. It is more likely that a drywell isolation valve, or valves, inadvertently left open would be necessary to exceed the limit. However, the licensee has presented several arguments to demonstrate that it is extremely unlikely that the drywell bypass leakage limit would be exceeded due to an inadvertently open drywell isolation valve. This is due to the large flow area necessary to exceed the allowable leakage value and the controls required by technical specifications to assure that the valves are closed.

The controls on the drywell isolation valve position are the same as the controls for similar valves in the primary containment. All automatic and remote manual isolation valves have position indication in the control room. Manual isolation valves and most check valves do not. The licensee stated in response to a staff question in the November 20, 1995, submittal, that automatic isolation valves that are not closed would either have an open indication (indicating that the valve is full open) or a dual indication (indicating the valve is somewhere between full open and full closed).

Each of the valves without position indication has a flow area of less than 8 inches. Calculations show that, even with all drywell isolation valves which are less than 8 inches in diameter in the open position, the bypass leakage design limit would not be exceeded. (However, the TSs limit would be exceeded.)

The Grand Gulf Nuclear Station drywell vacuum relief system has four sets of isolation valves isolating three 10 inch drywell penetrations (338, 339 and 340). One drywell penetration is isolated by two sets of drywell post LOCA vacuum relief subsystems in parallel, each consisting of one butterfly valve and one check valve. The other two penetrations are each isolated by drywell purge vacuum relief subsystems consisting of one butterfly valve and two check valves. The licensee provided the calculated effective A/√K values in the November 20, 1995, submittal for these penetrations. The A/√K values apply for forward flow (that is, from containment into the drywell) and are therefore conservative (i.e., lower) for flow in the opposite, or leakage, direction.

The licensee has shown that even with all four vacuum relief valves full open, the bypass leakage rate is less than the design limit A/√K of 0.9. (The TSs limit would be exceeded.)

The Grand Gulf Nuclear Station TSs require verification at least every 7 days that each vacuum breaker is closed. The position of the butterfly drywell isolation valves in each vacuum relief subsystem is indicated in the control room. Should a vacuum relief subsystem not be closed, the TSs provide only 4 hours to restore it to a closed position or begin a shut down.

River Bend Station does not have a vacuum relief system.

The licensee also assumed that one of the purge and exhaust penetration flow paths for Grand Gulf Nuclear Station and for River Bend Station is fully open in addition to other drywell bypass leakage equal to the TSs value. This is an A/√K value of approximately 0.7 for Grand Gulf Nuclear Station (see response to staff question 5(a) in the November 20, 1995, submittal). Thus, the design bypass leakage limit will not be exceeded in this case. For River Bend Station, the November 20, 1995, submittal gives an A/√K value of 1.0 ft^2. The design limit for River Bend Station is also 1.0 ft^2. Thus, a purge and exhaust penetration fully open plus other leakage at the TSs value would slightly exceed the design limit. However, the leakage would still be well below that necessary to exceed the actual containment failure limit.

The TSs of both plants require that these valves be maintained closed in MODES 1, 2, and 3 except under certain specified conditions when they are allowed to be opened under administrative control. This is verified at least every 31 days by using control room indication.

These examples demonstrate significant margin to the drywell bypass leakage limit for drywell isolation valves.

2.2.5 Air Locks and Equipment Hatch

The TSs require the drywell air lock to be leakage rate tested during every refueling outage. The test interval is currently 18 months. The licensee has proposed to change this interval to 24 months to accommodate longer operating cycles. As discussed in Sections 3.2 and 3.3, the staff finds this proposed change to be acceptable. In addition, the licensee has evaluated the effect of total loss of the drywell air lock seal for Grand Gulf Nuclear Station and has determined that the resulting leakage past the seals would not result in the drywell being unable to perform its safety function.

The TSs also require that the River Bend combination equipment hatch and personnel door assembly be leakage rate tested every refueling outage and that the seal pressure be verified every 7 days.

The equipment hatch at Grand Gulf has double compression seals and is leak tested under administrative controls before startup following each opening.

2.2.6 Electrical Penetrations

In discussions with the staff, the licensee provided a description of the electrical penetrations and discussed the likelihood of failure of an electrical penetration in such a manner as to provide a significant leakage path. The licensee concluded, based on the geometry of the penetration and the sealant used, that significant bypass leakage is highly unlikely. The sealant material is "very similar in practice to Portland cement" and is designed to resist accident pressure and temperature. In addition, the cable in the penetration limits the available flow path to some extent, even if there were no sealant.

As part of the rulemaking concerning the revision to 10 CFR Part 50, Appendix J, the staff examined the leakage behavior of primary containment electrical penetrations and found the operating experience was good enough to justify an increase in the leakage rate test interval from the two years specified in the previous rule to a maximum of 10 years (based on previous performance) under the new rule.

The staff therefore, concludes that the likelihood of significant leakage or failure of the electrical penetrations is very small.

2.2.7 Monitoring Leakage

The staff requested that the licensee consider a method of monitoring the drywell for significant leakage during operation. The licensee responded by proposing methods which provide a reasonable assurance that the TSs value of drywell bypass leakage will not be exceeded. A different method was proposed for each plant.

By letter dated December 12, 1995, the licensee agreed to perform a qualitative assessment of drywell leaktightness once per operating cycle at River Bend Station. The assessment will provide reasonable assurance that the drywell can perform its safety function; that is, remain OPERABLE. The licensee stated that the first assessment will be performed prior to Cycle 8; that is, sometime during Cycle 7 after the licensee has obtained the necessary data and performed calculations necessary for the assessment. Qualitative assessment means that it is not necessary to determine the value of drywell leakage if it can be assured that it is below a value that assures its safety function is maintained.

The licensee proposed to trend the River Bend drywell pressure versus containment pressure. Normal air inleakage into the containment requires periodic venting. The licensee proposes to trend drywell pressure changes vs. containment pressure changes as an indication of drywell bypass leakage. The staff finds this acceptable since the licensee has committed to provide a reasonable assurance of OPERABILITY. While the staff does not expect the test to be as accurate as the drywell bypass leakage test, the test will be able to detect gross leakage of a magnitude that would exceed the TSs limit.

By letter dated December 15, 1995, the licensee committed to assess drywell leaktightness at least once per operating cycle for Grand Gulf Nuclear Station. The assessment will provide a reasonable assurance that the drywell remains OPERABLE. The first assessment will be performed during Cycle 9, the first cycle for which the longer surveillance interval will be in effect. The licensee will perform the assessment using the purge compressors to cause a pressure change in the drywell. Although not as accurate as the TSs test, the test will indicate leakage at a level below the TSs limit.

2.2.8 Risk Considerations

Drywell performance plays a significant role in the risk analysis of the BWR/6. Radionuclides are released into the drywell atmosphere at vessel breach and during core concrete interaction. Early failure of the drywell is important because it would establish a pathway for radionuclides in the drywell to bypass the suppression pool. However, even with drywell failure or bypass, there still will be some reduction in the source term.

A rather simple analysis of the effect of drywell bypass on containment behavior can be obtained by using the analysis of the Grand Gulf Nuclear Station given in NUREG-1150 "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants". NUREG/CR-4551, Vol. 6, Rev. 1, Part 1, "Evaluation of Severe Accident Risks: Grand Gulf Unit 1, Main Report" provides some insight.

The conditional probability of drywell failure given core damage is 0.31. This is due to causes other than drywell bypass leakage. The probability of drywell bypass leakage in excess of the TSs value is taken to be zero. The mean probability of coincident early drywell failure and containment failure is 0.23. Therefore, there are some accidents which result in early drywell failure which do not result in early containment failure. However, for simplicity and conservatism, assume that the 0.31 conditional probability of drywell failure is also the probability of containment failure. Rather than using the probability of zero for drywell leakage, the staff conservatively assumed a value of 0.01 for the probability of a drywell bypass leakage path large enough to result in failure of the containment following a core damage event. This is a conservative estimate based on previous operating experience, the controls on penetrations discussed above and a test interval increase from 18 months to 10 years. Thus, to a first approximation, the conditional probability of drywell failure (including bypass) increases from 0.31 to 0.32. This is a small increase and would have only a small effect on risk.

Therefore, the staff considers the increase in risk due to the increase in the test interval from 18 months to 10 years to be acceptable.

Although the quantitative aspects of the above discussion are the result of calculations performed for the Grand Gulf Nuclear Station, the staff considers the conclusion for River Bend Station to be the same; that is, the fraction of risk contributed by drywell bypass leakage will be a small fraction of the total risk contribution considering all modes of drywell failure.

2.2.9 Staff Position

The staff reviewed the licensee's proposal to increase the test interval for drywell bypass leakage rate testing from 18 months to ten years. The staff finds this extension in the test interval to be acceptable. As discussed above, this is because of the demonstrated margin available due to the large amount of leakage necessary to exceed the containment design pressure, and the licensee's commitment to assess the drywell bypass leakage in order to maintain a reasonable assurance that the drywell remains OPERABLE.

3.0 PROPOSED TECHNICAL SPECIFICATION CHANGES

3.1 Drywell Bypass Leakage Rate Test Surveillance Interval Extension

A change to the surveillance frequency is proposed for the drywell bypass leakage test (Surveillance Requirement 3.6.5.1.1 for Grand Gulf Nuclear Station, Sureveillance Requirement 3.6.5.1.3 for River Bend Station) from 18 months to 10 years with an increased testing frequency required if performance degrades. An extension of the test interval by 12 months is permitted.

This change is discussed in Section 2.0 of this safety evaluation. The staff finds the licensee's proposal acceptable when modified by its commitment to perform an OPERABILITY assessment of the drywell at least once per cycle.

The licensee has proposed that following a drywell bypass test for which the leakage is greater than the drywell bypass leakage limit, tests will be required at an increased frequency of at least once every four years. Although this is not an Appendix J test, the decrease in the test interval upon failure of a drywell leakage rate test is consistent with the industry guidance approved in Regulatory Guide 1.163, September 1995, for a Type A primary containment leakage rate test.

Following two consecutive failed drywell leakage rate tests, the frequency will be returned to the current frequency of every refueling outage until two successful consecutive tests are performed.

3.2 Safety Evaluation on Drywell Air Lock Technical Specifications Changes for River Bend Station

3.2.1 Leakage rate surveillance is moved from the air lock LCO (3.6.5.2) to the drywell LCO (3.6.5.1).

The licensee proposes to move the air lock leakage rate surveillance requirement to the drywell LCO since excess air lock leakage will require actions for drywell inoperability. While this is different in format from the Improved Standard Technical Specifications, it is essentially an editorial change and the staff finds it acceptable.

3.2.2 Delete the requirement for the drywell air lock to meet a specific overall leakage rate limit.

The licensee states in the November 20, 1995, submittal that a drywell air lock leakage rate limit does not reflect the ability of the drywell to perform its safety function. This is not, however, the only purpose of this leakage requirement.

The drywell air lock leakage rate limit is intended as an indication of degradation. As such, however, it is not necessary as a TS value and the staff agrees that it may be removed from the TSs.

The TS value of allowable drywell air lock leakage for River Bend Station is 4.05 scfh. This is insignificant compared to the drywell leakage rate limit of approximately 46,000 scfm. Therefore, the staff finds this change acceptable.

3.2.3 Delete surveillance requirement 3.6.5.2.2 Note 1.

This note states:

> An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

The licensee states that the note "incorrectly implied that the drywell leakage limit could be exceeded due to an inoperable door without taking the actions for an inoperable drywell."

The staff finds this interpretation plausible and finds the licensee's proposal acceptable.

3.2.4 Change surveillance test interval for the drywell air lock leakage and the air lock interlock mechanism from 18 months to 24 months.

The staff finds this change acceptable because it is consistent with the guidance approved in Regulatory Guide 1.163, September 1995.

3.2.5 Relocate to the Bases the requirement in surveillance requirement 3.6.5.2.that the air lock leakage test at 3 psid be preceded by pressurizing the air lock to 19.2 psid.

The design basis post accident pressure is 3 psid. The peak pressure of 19.2 psid is not relevant to this test. Removing the requirement makes the River Bend Station TSs consistent with those of the other BWR/6's. The staff, therefore, finds this change acceptable.

3.2.6 Delete requirement that the drywell air lock seal leakage rate meet a specific leakage limit.

This change is acceptable since the ability of the drywell to perform its safety function is not dependent on the drywell air lock seals meeting a specific leakage limit. The overall drywell leakage limit provides the measure of the drywell's ability to perform its safety function.

The staff, therefore, finds this proposed change to be acceptable.

4.0 EVALUATION SUMMARY

The staff finds that the licensee's proposal to increase the drywell bypass leakage rate test interval from 18 months to 10 years is acceptable. This is based on the low increase in risk, the large margin for leakage, and the licensee's commitment to assess the drywell bypass leakage, and thereby assure operability, at least once every operating cycle.

The changes to the air lock TSs are acceptable in that they will add flexibility without decreasing safety for the reasons discussed above.

5.0 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.

6.0 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 (60 FR 62490). 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.

7.0 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.

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