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

January 23, 2009

Vice President, Operations
Entergy Operations, Inc.
River Bend Station
5485 US Highway 61N
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: CONTROL ROD NOTCH TESTING (TAC NO. MD9325)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 161 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 July 28, 2008.

The amendment (1) deletes TS surveillance requirement (SR) 3.1.3.2 and revises SR 3.1.3.3; (2) removes the reference to SR 3.1.3.2 from Required Action A.2 of TS 3.1.3, "Control Rod OPERABILITY"; (3) clarifies the requirement to fully insert all insertable rods for the limiting condition for operation in TS 3.3.1.2, "Source Range Monitor (SRM) Instrumentation," Required Action E.2; and (4) revises Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The changes are in accordance with NRC-approved TS Task Force (TSTF) traveler TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action."

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,

CFLyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures:
1. Amendment No. 161 to NPF-47
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

ENTERGY GULF STATES LOUISIANA, LLC

AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
License No. NPF-47

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated July 28, 2008, 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) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 161 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 and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License No. NPF-47 and Technical Specifications

Date of Issuance: January 23, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 161

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following pages of the Facility Operating License No. NPF-47 and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change.

Facility Operating License

Remove	Insert
-3-	-3-

Technical Specifications

Remove	Insert
1.0-27	1.0-27
1.0-28	1.0-28
3.1-7	3.1-7
3.1-9	3.1-9
3.3-11	3.3-11

(3) EOI, pursuant to the Act and 10 CFR Part 70, to receive, possess and to use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

EOI is authorized to operate the facility at reactor core power levels not in excess of 3091 megawatts thermal (100% rated power) in accordance with the conditions specified herein. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 161 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.

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-2 (continued)

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 23.8% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 23.8% RTP.

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----------------------------NOTE----------------------------- Not required to be performed until 12 hours after ≥ 23.8% RTP. --- Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 23.8% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 23.8% RTP, this Note allows 12 hours after power reaches ≥ 23.8% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 23.8% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 23.8% RTP.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

Once the unit reaches 23.8% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
--------------------------NOTE------------------------------ Only required to be met in MODE 1. --- Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour (plus the extension allowed by SR 3.0.2) interval, but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Perform SR 3.1.3.3 for each withdrawn OPERABLE control rod. AND A.3 Perform SR 3.1.1.1.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the Rod Pattern Control System (RPCS) 72 hours
B. Two or more withdrawn control rods stuck.	B.1 Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1 -----------NOTE---------- Inoperable control rods may be bypassed in RACS in accordance with SR 3.3.2.1.9, if required, to allow insertion of inoperable control rod and continued operation. ----------------------------- Fully insert inoperable control rod. AND	 3 hours (continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	DELETED	
SR 3.1.3.3	---------------------------NOTE-------------------------------- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RPCS. --- Insert each withdrawn control rod at least one notch.	31 days
SR 3.1.3.4	Verify each control rod scram time from fully withdrawn to notch position 13 is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Place reactor mode switch in the shutdown position.	1 hour
E. One or more required SRMs inoperable in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 161 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 July 28, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082120078, Reference 1), Entergy Operations, Inc. (the licensee), requested changes to the Technical Specifications (TSs) for River Bend Station, Unit 1 (RBS). The proposed changes would (1) delete TS surveillance requirement (SR) 3.1.3.2 and revise SR 3.1.3.3; (2) remove the reference to SR 3.1.3.2 from Required Action A.2 of TS 3.1.3, "Control Rod OPERABILITY"; (3) clarify the requirement to fully insert all insertable rods for the limiting condition for operation (LCO) in TS 3.3.1.2, "Source Range Monitor (SRM) Instrumentation," Required Action E.2; and (4) revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension.

The changes are in accordance with U.S. Nuclear Regulatory Commission (NRC)-approved TS Task Force (TSTF) traveler TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action." The NRC issued a *Federal Register* notice on November 13, 2007 (72 FR 63935) announcing the availability of this TS improvement through the consolidated line item improvement process.

TSTF-475 revised the reference Standard Technical Specifications (STS) by: (1) revising the frequency of SR 3.1.3.2, notch testing of each fully withdrawn control rod, from 7 days after the control rod is withdrawn and THERMAL POWER is greater than the Low Power Setpoint (LPSP) of the Rod Worth Minimizer (RWM) to "31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM," and (2) revising Example 1.4-3 in Section 1.4, "Frequency," to clarify that the 1.25 surveillance test interval extension in SR 3.0.2 is applicable to time periods discussed in NOTES in the "SURVEILLANCE" column in addition to the time periods in the "FREQUENCY" column. NUREG-1433 (Reference 2) is the STS for General Electric plants like RBS, which is a boiling-water reactor (BWR)-6.

The licensee stated in Section 2.0 of Attachment 1 to its application that it is not proposing any variations or deviations from the (1) applicable TS changes described in the modified TSTF-475, Revision 1 and (2) NRC staff's model safety evaluation dated November 13, 2007. The licensee also stated that the justifications presented in the TSTF and the NRC staff safety evaluation for the TSTF are applicable to RBS.

The purpose of the surveillances is to confirm control rod insertion capability which is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. Control rods and the control rod drive (CRD) mechanism (CRDM), by which the control rods are moved, are components of the CRD system (CRDS), which is the primary reactivity control system for the reactor. By design, the CRDM is highly reliable, with a tapered design of the index tube which is conducive to control rod insertion.

A stuck control rod is an extremely rare event, and industry review of plant operating experience did not identify any incidents of stuck control rods while performing a rod notch surveillance test.

The purpose of these revisions is to reduce the number of control rod manipulations and, thereby, reduce the opportunity for reactivity control events.

The purpose of the change to Example 1.4-3 in Section 1.4, "Frequency," is to clarify the applicability of the 25 percent allowance of SR 3.0.2 to time periods discussed in NOTES in the "SURVEILLANCE" column as well as to time periods in the "FREQUENCY" column.

2.0 REGULATORY EVALUATION

In Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36), the Commission established its regulatory requirements related to the content of the TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls.

As stated in 10 CFR 50.36(c)(2)(i), LCOs are "the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications ..." The remedial actions in the TSs are specified in terms of LCO conditions, required actions, and completion times (CTs), or allowed outage times (AOTs), to complete the required actions. When an LCO is not being met, the CTs specified in the TSs are the time allowed in the TSs for completing the specified required actions. The conditions and required actions specified in the TSs must be acceptable remedial actions for the LCO not being met, and the CTs must be a reasonable time for completing the required actions while maintaining the safe operation of the plant.

As required by 10 CFR 50.36(c)(2)(ii), an LCO must be included in TSs for any item meeting one of the following four criteria:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Those items that do not fall within or satisfy any of the above criteria are not required to be included in the TSs.

As required by 10 CFR 50.36(c)(3), SRs are the requirements related to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

As required by 10 CFR 50.36(c)(5), administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criterion (GDC) 29, "Protection against anticipated occurrence," requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences. The design relies on the CRDS to function in conjunction with the protection systems under anticipated operational occurrences, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRDS provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during anticipated operational occurrences. Meeting the requirements of GDC 29 for the CRDS prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during anticipated operational occurrences. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

The design of the control rods in the reactivity control system is not being changed by the proposed amendment. The SR to demonstrate operability of the control rods by inserting each fully withdrawn control rod at least one notch is proposed to be deleted and another SR would be revised, which is to combine the notch testing of both the fully withdrawn and partially withdrawn control rods into a single surveillance. The SR being deleted would be removed from the remedial actions for an inoperable control rod. Although the design of the control rods is not being changed, the frequency of notch testing the fully withdrawn control rods is being extended to 31 days and this reduced frequency of surveillance may reduce the reliability required by GDC 29 of these control rods.

3.0 TECHNICAL EVALUATION

3.1 Proposed Changes to TSs

In its application, the licensee proposed the following changes to TS 3.1.3, "Control Rod Operability, and TS 3.3.1.2, "Source Range Monitor (SRM) Instrumentation," of the RBS TSs:

1. Delete SR 3.1.3.2 by replacing the surveillance text in SR 3.1.3.2 by the word "deleted,"

2. Revise SR 3.1.3.3 by deleting the word "partially" and extending the time to perform the surveillance to 31 days,

3. Remove the reference to SR 3.1.3.2 from Required Action A.2 for the condition of one withdrawn control rod stuck, and

4. Add the word "fully" to Required Action E.2 of TS 3.3.1.2 to state "Initiate action to fully insert all insertable control rods ..."

To be clear that the allowance of the 1.25 surveillance test interval extension in SR 3.0.2 is applicable to time periods discussed in notes for SRs (e.g., in SR 3.1.3.3), the Example 1.4-3 in TS Section 1.4, "Frequency," is also revised to add the phrase in two places: "(Plus the extension allowed by SR 3.0.2)."

Although the licensee proposed the deletion of SR 3.1.3.2, SR 3.1.3.2 will remain in the TSs with the surveillance text replaced by the word "deleted." The reason for this is to not have to re-number SRs 3.1.3.3 through 3.1.3.5 and, therefore, change the reference to SR 3.1.3.4 in the second note to TS Table 3.1.4-1, "Control Rod Scram Times." Therefore, the identified re-numbering of (1) SRs 3.1.3.3 through 3.1.3.5 and (2) SR 3.1.3.4 in the second note of TS Table 3.1.4-1 in TSTF-475, Revision 1, are not necessary and were not proposed by the licensee. Thus, the proposed changes are in accordance with the NRC-approved TSTF-475, Revision 1, except for the re-numbering of certain SRs.

3.2 Background

The CRDS is the primary reactivity control system for the reactor. The CRDS, in conjunction with the Reactor Protection System, provides the means for the reliable control of reactivity changes to ensure under all conditions of normal operation, including anticipated operational occurrences that specified acceptable fuel design limits are not exceeded. Control rods are components of the CRDS that have the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRDS.

The CRDS consists of a CRDM, by which the control rods are moved, and a hydraulic control unit (HCU) for each control rod. The CRDM is a mechanical hydraulic latching cylinder that positions the control blades. The CRDM is a highly reliable mechanism for inserting a control rod to the full-in position. The collet piston mechanism design feature ensures that the control rod will not be inadvertently withdrawn. This is accomplished by engaging the collet fingers, mounted on the collet piston, in notches located on the index tube. Due to the tapered design of the index tube notches, the collet piston mechanism will not impede rod insertion under normal insertion or scram conditions.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD which houses the collet mechanism which consist of the locking collet, collet piston, collet return spring and an unlocking cam. The collet mechanism provides the locking/unlocking mechanism that allows the insert/withdraw movement of the control rod. The CRT has three primary functions: (a) to carry the hydraulic unlocking pressure to the collet piston, (b) to provide an outer cylinder, with a suitable wear surface for the metal collet piston rings, and (c) to provide mechanical support for the guide cap, a component which incorporates the cam surface for holding the collet fingers open and also provides the upper rod guide or bushing.

The control rods are required by TS 3.1.3 to be operable in Modes 1 (power operation) and 2 (startup), when the reactor is critical. The requirements on the operability of the control rods and the design of the control rods are not being changed by the amendment.

The operability of the control rods is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the rod moved the one notch. The control rod may then be returned to its original position. This action ensures that the control rod is not stuck and is free to be inserted on a scram signal. It does not demonstrate that the control rod would in fact be inserted into the core on a scram signal or how fast the rod would be inserted if there was a signal to scram the reactor. The control rods for a BWR are below the reactor vessel and are inserted up into the core for reactivity control.

Fully withdrawn control rods are tested in accordance with SR 3.1.3.2 on a frequency of 7 days. The partially inserted control rods are tested in accordance with SR 3.1.3.3 on a 31-day frequency. This frequency is based on the potential power reduction required to allow the control rod movement, which affects the reactivity in the core and the power level.

The proposed change would combine SRs 3.1.3.2 (fully withdrawn rods) and 3.1.3.3 and (partially withdrawn rods) into a single SR 3.1.3.3 (SR 3.1.3.2 remains, but its text is replaced by the word "deleted") to move all the control rods at least one notch. In deleting SR 3.1.3.2 and revising SR 3.1.3.3 to remove the word "partially" so that the requirement is to move all withdrawn control rods (fully withdrawn or partially withdrawn) at least one notch., the SRs on the fully and partially withdrawn control rods is not being changed. What is being changed by combining the two SRs into a single SR for notch testing is the following:

1. The frequency of SR 3.1.3.2 for fully withdrawn controls rods is being extended from the current 7 days to 31 days. Therefore, the number of times a fully withdrawn control rod would be tested by inserting the rod at least one notch would be reduced.

2. The Note for the combined SR 3.1.3.3 for both the fully and partially withdrawn control rods would be revised such that the time to perform the SR after the control rod was fully or partially withdrawn would be extended from (1) 8 days and 18 hours for the fully withdrawn control rod and (2) 38 days and 18 hours for the partially withdrawn control rod to 31 days for both withdrawn control rods.

The Notes specifying the time to perform the current SRs 3.1.3.2 (fully withdrawn control rods) and 3.1.3.3 (partially withdrawn control rods) are timed from when the control rods were last withdrawn (fully or partially) concurrent with reactor thermal power greater than the LPSP of the rod pattern control system (RPCS). This allowance in the Notes that the thermal power is greater than the LPSP of the RPCS is not being changed by this amendment.

3.3 TSTF-475, Revision 1

The NRC staff previously reviewed the following information provided by the TSTF to support the staff's review and approval of TSTF-475, Revision 1. Specifically, the following documents were reviewed during the NRC staff's evaluation:

- TSTF letter TSTF-04-07 (Reference 3) - Provided a description of the proposed changes in TSTF-475 that changes the weekly rod notch frequency to monthly and clarify the applicability of the 25 percent allowance in Example 1.4-3.

- TSTF letter TSTF-06-13 (Reference 4) - Provided responses to NRC staff request for additional information (RAI) on (1) industry experience with identifying stuck rods, (2) tests that would identify stuck rods, (3) continued compliance with General Electric (GE) Services Information Letter (SIL) No. 139 (Reference 5), (4) industry experience on collet failures, and (5) applying the 25 percent grace period to the 31-day control rod notch SR test frequency.

- Boiling Water Reactor Owners Group (BWROG) letter BWROG-06036 (Reference 6) – Provided the GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," in which CRD notching frequency and CRD performance were evaluated.

- TSTF letter TSTF-07-19 (Reference 7) - Provided response to NRC staff RAI on CRD performance in Control Cell Core (CCC)-designed plants, including TSTF-475, Revision 1.

The NRC staff approved TSTF-475, Revision 1, which revised the TS SR 3.1.3.2, "Control Rod Operability," in the STS from 7 days to 31 days based on the following: (1) slow crack growth rate of the CRT; (2) the improved CRT design; (3) a higher reliable method (scram time testing) to monitor CRD scram system functionality; (4) GE chemistry recommendations; and (5) no known CRD failures have been detected during the notch testing exercise. The NRC staff concluded that the changes would reduce the number of control rod manipulations thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the extremely high reliability of the CRDS.

The following paragraphs describe the bases for the NRC staff's approval of TSTF-475, Revision 1.

According to the BWROG, at the time of the first CRT crack discovery in 1975, each partially or fully withdrawn operable control rod was required to be exercised one notch at least once each week. It was recognized that notch testing provided a method to demonstrate the integrity of the CRT. Control rod insertion capability was demonstrated by inserting each partially or fully withdrawn control rods at least one notch and observing that the control rod moved. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal.

It was determined that during scrams, the CRT temperature distribution changes substantially at reactor operating conditions. Relatively cold water moves upward through the inside of the CRT and exits via the flow holes into the annulus on the outside. At the same time, hot water from the reactor vessel flows downward on the outside surface of the CRT. There is very little mixing of the cold water flowing from the three flow holes into the annulus and the hot water flowing downward. Thus, there are substantial through-wall and circumferential temperature gradients during scrams which contribute to the observed CRT cracking.

Subsequently, many BWRs have reduced the frequency of notch testing for partially withdrawn control rods from weekly to monthly. The notch test frequency for fully withdrawn control rods are still performed weekly. The change for partially withdrawn control rods was made because of the potential power reduction required to allow control rod movement for partially withdrawn control rods, the desire to coordinate scheduling with other plant activities, and the fact that a large sample of control rods are still notch tested on a weekly basis. The operating experience related to the changes in CRD performance also provided additional justification to reduce the notch test frequency for the partially withdrawn control rods.

To support its position to reduce the CRD notch testing frequency, the BWROG provided plant data and a GE Nuclear Energy report entitled, "CRD Notching Surveillance Testing for Limerick Generating Station." The GE report provided a description of the cracks noted on the original design CRT surfaces. These cracks, which were later determined to be intergranular, were generally circumferential, and appeared with greatest frequency below and between the cooling water ports, in the area of the change in wall thickness. Subsequently, cracks associated with residual stresses were also observed in the vicinity of the attachment weld. Continued circumferential cracking could lead to 360 degree severance of the CRT that would render the CRD inoperable which would prevent insertion, withdrawal or scram. Such failure would be detectable in any fully or partially withdrawn control rod during the surveillance notch testing required by the TSs. To a lesser degree, cracks have also been noted at the welded joint of the interim design CRT but no cracks have been observed in the final improved CRT design. Neither the BWROG nor the NRC staff was able to find evidence of a collet housing failure since 1975. To date, operating experience data shows no reports of a severed CRT at any BWR. No collet housing failures have been noted since 1975. On a numerical basis for instance, based on a BWROG assumption that there are 137 control rods for a typical BWR/4 and 193 control rods for a typical BWR/6, the yearly performance would be 6590 rod notch tests for a BWR/4 plant and 9284 for a BWR/6 plant. For example, if all BWRs operating in the U.S. are taken into consideration, the yearly performances of rod notch data would translate into approximately 240,000 rod notch tests without detecting a failure.

In addition, the intergranular stress-corrosion cracking (IGSCC) crack growth rates were evaluated at Limerick Generating Station, using GE's PLEDGE model with the assumption that the water chemistry condition is based on GE recommendations. The model is based on fundamental principles of stress-corrosion cracking which can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. It was determined that the additional time of 24 days represented an additional 10 mils of growth in total crack length. The small difference in growth rate would have little effect on the behavior between one notch test and the next subsequent test. Therefore, from the materials perspective based on low crack growth rates, a decrease in the notch test frequency would not affect the reliability of detecting a CRDM failure due to crack growth.

The BWR scram system has extremely high reliability. In addition to notch testing, scram time testing can identify failure of individual CRD operation resulting from IGSCC-initiated cracks and mechanical binding. Unlike the CRD notch tests, these single rod scram tests cover the other mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator, as well as operation of the control rods. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod.

Also, the CRD drives, the hydraulic control units (HCUs) of the drives, and the control rods are tested during refueling outages, approximately every 18-24 months. Based on the data collected during the preceding cycle of operation, selected control rod drives are inspected and, as required, their internal components are replaced. Therefore, increasing the CRD notch testing frequency to monthly would have very minimal impact on the reliability of the scram system.

3.4 Evaluation of Proposed Amendment

<u>Delete SR 3.1.3.2 for Fully Withdrawn Control Rods</u>

As discussed in Sections 3.1 and 3.2 of this safety evaluation, the fully withdrawn control rods are inserted at least one notch in accordance with SR 3.1.3.2 on a 7 day frequency and the partially withdrawn control rods, in accordance with SR 3.1.3.3 on a 31 day frequency. The proposed change revises the frequency of SR 3.1.3.2 from 7 days to 31 days, but keeps the frequency for SR 3.1.3.3 unchanged. As a result, the frequency for testing of all withdrawn controls rods will be 31 days. Hence, the existing SRs 3.1.3.2 and 3.1.3.3 are proposed to be combined with an extended frequency of 31 days, the frequency for SR 3.1.3.3. The notes for SRs 3.1.3.2 and 3.1.3.3 have a time specified based on the frequency for the surveillance. This time would be changed to 31 days for the revised SR 3.1.3.3 to be the same as the frequency of the surveillance.

This change in the frequency of the notch insertion surveillances is shown in the table below. The change is shown with the overall number of control rod notch insertion surveillances for a typical BWR in a year. The data is given for both a BWR/4 and BWR/6, even though RBS is a BWR/6. The information comes from GE SIL 39 (Reference 5), the GE report GE-NE-0000-0024-9859 R0 (Reference 8), and the four documents listed in Section 3.3 of this safety evaluation.

The purpose of these two surveillances is to confirm control rod insertion capability (i.e., that the control rods can move). However, a stuck control rod is an extremely rare event. The CRDM, by design, is highly reliable and the tapered design of the index tube is conducive to control rod insertion. A review of industry operating experience did not identify any incidents of stuck control rods identified via performance of a rod notch surveillance. The following table illustrates the impact of the proposed change on the overall number of control rod notch surveillances performed in a year for a typical BWR reactor. It is assumed that there are 137 control rods in the typical BWR/4 and 193 control rods in a typical BWR/6. Of these controls rods, approximately 90 percent are fully withdrawn during power operation.

Surveillance Requirement	Frequency of Surveillance	Yearly Performance	
		BWR/4	BWR/6
Current SR 3.1.3.2 (Fully Withdrawn Control Rod)	7 days	6429	9057
Current SR 3.1.3.3 (Partially Withdrawn Control Rod)	31 days	161	227
Total		6590	9284
Proposed SR 3.1.3.2 (All Withdrawn Control Rods)	31 days	1613	2272
Total		1613	2272

Given the demonstrated reliability of the CRDMs, the NRC staff concludes that the performance of weekly notch testing of fully withdrawn control rods to confirm the capability of inserting such rods is not necessary.

The large number of tests that would still be performed will provide a very high confidence that any problems with the system would be identified. Should a control rod be determined to be stuck, TS 3.1.3 Required Action A.3 continues to require that a notch test of each withdrawn control rod be performed within 24 hours of the discovery of the stuck rod. This requirement will ensure that a generic problem does not exist.

The reduction in the number of control rod positioning steps prevents unnecessary control rod manipulations and has a two fold benefit. First, it will reduce the duty on the reactor manual control system and CRD hardware, which will improve equipment reliability because it reduces the number of control rod manipulations. Second, it reduces the number of potential reactivity control errors that could occur, because it reduces the number of operator actions. The potential effects of reducing the number of notch tests are far outweighed by the benefits of (1) reducing undue equipment wear, (2) reducing unnecessary burden on reactor operators and (3) reducing the potential for mispositioning events which accompanies any control rod manipulation.

The safety function of the control rods, in the event of a design-basis accident (DBA) or transient, is to provide the primary means of rapid reactivity control (i.e., scram). Notch testing does not specifically ensure this safety function, but rather it only verifies that the rod has freedom of movement (i.e., capable of scramming by inference of the control rod movement). The assurance that control rods are capable of scramming is provided by the required surveillances in TS 3.1.4, "Control Rod Scram Times," and TS 3.1.5, "Control Rod Scram Accumulators." The proposed change is limited to only the notch testing surveillance and, as such, the TS 3.1.4 and TS 3.1.5 surveillances will continue to ensure that the performance of the control rods in the event of a DBA or transient meets the assumptions used in the safety analyses.

The TS 3.1.4 and TS 3.1.5 surveillances are more likely to identify issues which may affect the ability of the control rods to perform their safety function, such as (1) fuel channel bowing, which occurs nearer to the center of the fuel channel and would not be identified by notch testing of full out rods, or (2) mispositioning of manual isolation valves on the HCUs causing failure to scram of individual control rods, which would most likely occur during maintenance activities and would be apparent during scram time testing performed prior to or during the return to operation (as required by SRs 3.1.4.3 and 3.1.4.4). Failure mechanisms expected to be found via notch testing would be more gradual in nature, such as debris (i.e., crud buildup) within the CRDM affecting normal operation of the control rods. The NRC staff concludes that the proposed frequency for notch testing each fully withdrawn control rod every 31 days is more than adequate to detect such gradual changes. The frequency for notch testing the partially withdrawn rods is not being changed.

Revising the frequency for notch testing fully withdrawn control rods will have the indirect effect of reducing the number of coupling checks performed in accordance with the existing SR 3.1.3.5, which requires coupling checks be performed any time a control rod is fully withdrawn. However, coupling integrity continues to be assured, because of the improbability of a control rod becoming decoupled when it has not been moved.

Another use of notch testing of fully withdrawn control rods is to identify collet/flange tube cracking. This cracking is discussed in GE SIL No. 139 (Reference 5). GE, the control rod drive manufacturer, does not specify any particular preventative maintenance frequency for CRDMs. However, GE recommended in 1975, as part of SIL No. 139, that each control rod drive mechanism be exercised weekly to detect a failure in the collet housing region of the control rod drive flange tube. A collet housing failure could result in the inability to insert, withdraw, and/or scram a control rod. SR 3.1.3.2 ensured compliance with the SIL No. 139 recommendation. However, GE has since evaluated the acceptability of the proposed change for Limerick Generating Station and the results of the evaluation are documented in GE Nuclear Energy Report GE-NE-0000-0024-9858 R0 (Reference 8). The GE evaluation concluded that extending the control rod notch testing frequency for fully withdrawn control rods from 7 days to 31 days would not compromise the material condition or reliability of the CRD system. Furthermore, the evaluation concluded that monthly control rod notch testing was adequate to detect collet housing failures given the slow collet housing crack growth rate.

In summary, the CRDs and CRDMs are extremely reliable systems and, as such, reducing the number of control rod notch tests on fully withdrawn rods will not significantly impact the likelihood of detecting an inoperable control rod. If an inoperable control rod is detected, existing action requirements will ensure prompt action is taken to ensure there is not a generic problem. Other surveillances (e.g., SR 3.1.4.2) are routinely performed to ensure the safety function of the control rods to scram in the event of a DBA or transient meets the assumptions used in the safety analyses. As such, potential effects of reducing the number of notch tests are far outweighed by the benefit of reducing undue burden on reactor operators, reducing the potential for mispositioning events which accompanies any control rod manipulation, and reducing undue equipment wear.

The licensee has proposed to delete SR 3.1.3.2 for the fully withdrawn control rods and to remove the word "partially" from SR 3.1.3.3 for the partially withdrawn control rods. By removing the word "partially," SR 3.1.3.3 will apply to any withdrawn control rod. The revised SR 3.1.3.3 would then apply to both fully withdrawn and partially withdrawn control rods.

Therefore, based on the NRC staff's review of the above proposal, the NRC staff concludes that (1) the deletion of SR 3.1.3.2 by combining SRs 3.1.3.2 (fully withdrawn controls) and 3.1.3.3 (partially withdrawn control rods) into one surveillance for any withdrawn control rod, and the extension of the frequency to 31 days for the fully withdrawn control rods is acceptable.

Deleting SR 3.1.3.2 from TS 3.1.3 Required Action A.3

The licensee proposes to delete SR 3.1.3.2, and replace it with the word "deleted." Deleting SR 3.1.3.2 from Required Action A.2 does not delete any requirements from the TSs, because SR 3.1.3.3 now applies to both fully withdrawn and partially withdrawn control rods.

Based on the above, the NRC staff concludes that the proposed change is acceptable.

Adding the Word "fully" to TS 3.3.1.2 Required Action E.2

Regarding the change to TS 3.3.1.2, Required Action E.2, the requirement to insert control rods is meant to require control rods to be fully inserted. Other similar required actions in STSs also require the control rods to be fully inserted and the TSTF-475, Revision 1, contains the word "fully" in this required action. The addition of the word "fully" was not shown in TSTF-475, Revision 1, because, as stated in the previous sentence, the word "fully" was already in the STS for 3.3.1.2 and did not have to be added. Based on this, the NRC staff concludes that the proposed addition of the word "fully" clarifies the action and does not add any new requirement to the TSs.

Based on the above, the NRC staff also concludes that the proposed change is acceptable and, therefore, meets 10 CFR 50.36.

Adding a Phrase to Example 1.4-3

Regarding the change to Example 1.4-3 in Section 1.4, “Frequency,” this change makes it clear that the 1.25 provision in SR 3.0.2 is equally applicable to time periods specified in the “FREQUENCY” column and in Notes in the “SURVEILLANCE” column. This change to Example 1.4-3 is linked to TSTF-475 since the newly re-numbered SR 3.1.3.2 contains a 31 day time period in both the “SURVEILLANCE” column and in the “FREQUENCY” column, and the revised Example makes it clear that the 1.25 provision is equally applicable to both of these 31 day periods in SR 3.1.3.2. This change is proposed to be consistent with the definition of “specified Frequency” provided in the second paragraph of Section 1.4. This paragraph states:

> The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

As made clear in the second sentence above, the "specified Frequency" includes time periods discussed in Notes in the "Surveillance" column, in addition to time periods listed in the "Frequency" column. Therefore, the provisions of SR 3.0.2 (which permit a 25 percent grace period to facilitate surveillance scheduling and avoid plant operating conditions that may not be suitable for conducting the test) also apply to the time periods listed in Notes in the "SURVEILLANCE" column. This is because SR 3.0.2 states that "The *specified Frequency* (emphasis added) for each SR is met if the Surveillance is performed within 1.25 times the interval specified...."

Therefore, the licensee proposes to revise Example 1.4-3 to be consistent with the above statements. The example currently explicitly recognizes that the 25 percent extension allowed by SR 3.0.2 is applicable to the time period listed in the "FREQUENCY" column, but it does not explicitly recognize that the SR 3.0.2 extension is applicable to the time period listed in the NOTE in the "SURVEILLANCE" column. The change to the Example provides this explicit recognition by copying the phrase "(plus the extension allowed by SR 3.0.2)" in two additional portions of the discussion for this Example.

Based on the above, the NRC staff concludes that the proposed addition of the phrases to Example 1.4-3 of the RBS TSs meets the requirements of 10 CFR 50.36 and is acceptable.

3.5 TS Bases

The licensee committed in its submittal to establishing TS Bases consistent with TSTF-475, Revision 1. The NRC staff has no objections.

3.6 Conclusion

Based on its evaluation in the previous section of this safety evaluation, of the proposed changes to the TSs in this amendment, the NRC staff finds these changes meet the requirements of 10 CFR 50.36 and are acceptable. The NRC staff also concludes that the proposed TS revisions will have a minimal effect on the high reliability of the CRDS, while reducing the opportunity for potential reactivity events. Thus, the plant continues to meet the requirements of GDC 29. Since the amendment request meets the requirements of GDC 29 and 10 CFR 50.36, the NRC staff concludes that the proposed changes are acceptable.

4.0 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.

5.0 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 a surveillance requirement. 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 published in the *Federal Register* on November 4, 2008 (73 FR 65690). 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.

6.0 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.

7.0 REFERENCES

1. Entergy Operations, Inc. to U.S. Nuclear Regulatory Commission, "License Amendment Request, Application for Technical Specification Changes Using the Consolidated Line Item Improvement Process (CLIIP)," dated July 28, 2008, ADAMS Accession No. ML082120078.

2. U.S. Nuclear Regulatory Commission, "Standard Technical Specifications General Electric Plants," NUREG-1434, Vol. 1 and 2, Revision 3, August 31, 2003, ADAMS Accession Nos. ML032050129 and ML032050135.

3. Technical Specifications Task Force to U.S. Nuclear Regulatory Commission, "TSTF-475 Revision 0, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," TSTF-04-07, dated August 30, 2004, ADAMS Accession No. ML042520035.

4. Technical Specifications Task Force to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0," TSTF-06-13, dated July 3, 2006, ADAMS Accession No. ML061840342.

5. General Electric (GE) Service Information Letter (SIL) No. 139, "Control Rod Drive Collet Retainer Tube Cracking," dated July 18, 1975, including supplements.

6. BWR Owners Group to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0," BWROG-06036, dated November 16, 2006, with Enclosure of the GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," dated November, 2006, ADAMS Accession No. ML063250258.

7. Technical Specifications Task Force to U.S. Nuclear Regulatory Commission, “Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0, ‘Control Rod Notch Testing Frequency and SRM Insert Control Rod Action,’ dated February 28, 2007," (TSTF-475 Revision 1 is an enclosure) TSTF-07-19, dated May 22, 2007, ADAMS Accession No. ML071420428.

8. GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," GE-NE-000-0024-9858 R0, dated February 2004.

Principal Contributors: R. Grover
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Date: January 23, 2009

January 23, 2009

Vice President, Operations
Entergy Operations, Inc.
River Bend Station
5485 US Highway 61N
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: CONTROL ROD NOTCH TESTING (TAC NO. MD9325)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 161 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 July 28, 2008.

The amendment (1) deletes TS surveillance requirement (SR) 3.1.3.2 and revises SR 3.1.3.3; (2) removes the reference to SR 3.1.3.2 from Required Action A.2 of TS 3.1.3, "Control Rod OPERABILITY"; (3) clarifies the requirement to fully insert all insertable rods for the limiting condition for operation in TS 3.3.1.2, "Source Range Monitor (SRM) Instrumentation," Required Action E.2; and (4) revises Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The changes are in accordance with NRC-approved TS Task Force (TSTF) traveler TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action."

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,

/RA/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures:
1. Amendment No. 161 to NPF-47
2. Safety Evaluation

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OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	DIRS/ITSB/BC	OGC	NRR/LPL4/BC	NRR/LPL4/PM
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