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

December 19, 2008

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

SUBJECT: RIVER BEND STATION, UNIT 1 - REQUEST FOR ADDITIONAL INFORMATION FOR MAIN TURBINE BYPASS SYSTEM LICENSE AMENDMENT REQUEST (TAC NO. MD7966)

Dear Sir or Madam:

In your application dated January 25, 2008 (RBG-46690), Entergy Operations, Inc. (the licensee) proposed an amendment to the Technical Specifications (TSs) for the River Bend Station, Unit 1. In the application, changes are proposed to TS 3.7.5, "Main Turbine Bypass System [MTBS]," to allow the plant to operate with the MTBS inoperable by revising reactor operating limits and remedial actions.

Enclosed is a request for additional information that is needed for the Nuclear Regulatory Commission (NRC) staff to complete its review of your application. These questions were forwarded via electronic mail and discussed with your staff so that your staff would understand what information is needed by the NRC staff and could provide the NRC staff a schedule by which the information would be submitted. Any differences between the enclosed questions and the electronic mail to your staff are editorial, deletion of questions that duplicated other questions, or revised questions to clarify what is needed.

In telephone conference calls on December 4 and 18, 2008, your staff agreed to submit the responses by early February 2009 with a discussion on the responses before January 31, 2009. This schedule will not allow the NRC staff to complete its evaluation within its goal of completing the licensing action within 1 year of the application date. However, your staff agreed that it is necessary to complete this review now that it is clear what is needed by the NRC staff to complete its review and that the revised schedule will allow the review to be completed as close to the 1-year performance metric as practicable without stopping the review. During the calls, your staff did not identify any proprietary information beyond what the staff identified in the enclosed questions. If you have any questions, please contact me at 301-415-1307, or via e-mail at jack.donohew@nrc.gov.

Sincerely,

Jack Donohew

Jack Donohew, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosure

cc w/encl: See next page

River Bend Station (10/9/2008)

cc:

Ms. H. Anne Plettinger
3456 Villa Rose Drive
Baton Rouge, LA 70806

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OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

MAIN TURBINE BYPASS SYSTEM LICENSE AMENDMENT REQUEST

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

The request for additional information (RAI) pertains to the application dated January 25, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML080440293), by Entergy Operations, Inc. (the licensee), to revise Technical Specification (TS) 3.7.5, "Main Turbine Bypass System [MTBS]," to allow River Bend Station, Unit 1, to operate with the MTBS inoperable by revising (1) reactor operating limits and (2) TS 3.7.5 remedial actions.

Based on its review of the application, the Nuclear Regulatory Commission (NRC) staff requests additional information listed below in order to complete its review. For the questions listed under RAI #6, proprietary information, identified within square brackets as "[xxxxx]," was only provided to the licensee by email.

1. Specify the "certain accidents" that will be affected by the proposed change. Describe the rationale for choosing the accidents such that it is clear that the most limiting scenarios were chosen to determine the penalties for the operating limits. Specify the NRC-approved code(s) that are utilized for the chosen accidents.

2. Describe the out-of-service analysis, results, and effect on the revised reactor operating limits.

3. Describe the expected differences between the cycle-specific analysis with the proposed changes and the most recent cycle-specific analysis. Describe what additional information will be included in the core operating limits report (COLR).

4. In the licensee's draft response to Question 1, it appears that the NRC-approved topical report to be used to re-calculate the average planar linear heat generation rate (APLHGR), minimum critical power ratio (MCPR), and linear heat generation rate (LHGR) limits for an inoperable MTBS is stated to be XN-NF-80-19(P), Volume 3, Revision 2. There are four volumes of XN-NF-80-19(P). Volume 3, Revision 2 appears to address abnormal operational occurrences (AOOs), but does not address accidents. Volume 4, for example, appears to address AOOs and accidents. Also, since the APLHGR limit is derived from the emergency core cooling system (ECCS) analysis for loss-of-coolant accidents (LOCAs), it appears that Volume 3 would be inappropriate for calculating the APLHGR limit.

   For the APLHGR, MCPR, and LHGR limits for an inoperable MTBS, provide the volume of XN-NF-80-19(P) that is being used to calculate each limit.

Enclosure

5. In each volume of XN-NF-80-19(P), there is the following: (1) the conditions and limitations on the use of the topical report in the NRC safety evaluation (SE) approving the use of the topical report, which is enclosed in the topical report, and (2) a discussion of the appropriate AOOs and/or accidents or both to be addressed in calculating the core limits. Discuss how (1) the conditions and limitations in the appropriate NRC SE are met, (2) the affected AOO and accident analyses are identified, and (3) appropriate AOOs and accidents are being considered consistent with the discussions in the applicable volume of XN-NF-80-19(P). Provide a list of the affected safety analyses. Provide justification for any AOOs and accidents specified in the applicable volume of XN-NF-80-19(P) that are not re-analyzed for the MTBS being inoperable.

6. To support the review of a license amendment for a power uprate, the NRC staff conducted a review of the applicability of the NRC-approved AREVA suite of nuclear design and transient analysis methods. As part of this review, an audit of the AREVA codes was conducted. The result of the audit is that the staff needs additional information for clarification on the use of the AREVA suite of methods. Until this is resolved with AREVA, these questions are being requested on plant-specific applications that use this suite of methods. These questions for clarification of the River Bend application are the following:

    6.1 Verify that the upstream transient COTRANSA2 analysis: (1) includes the 110 percent integral thermal power multiplier, (2) biases all relevant input parameters to the limiting values allowable by TSs as appropriate, (3) biases non-TS-controlled input parameters to the most conservative value based on their associated uncertainty, and (4) is representative of the limiting plant configuration allowable for equipment out of service in the TSs.

    6.2. If COTRANSA2 or another one-dimensional code is used to determine the transient reactor power, please describe how appropriate axial planar average fuel rod parameters are determined for the analysis. This discussion should address: gap conductance, thermal conductivity, pellet size, and heat capacity. Justify that these parameters are acceptably accurate or conservative. Demonstrate that the conservatism in the COTRANSA2 analysis is sufficient to bound any bias in the transient peak heat flux calculation as a result of known [xxxxx] for ATRIUM-10 fuel.

    6.3. For the transient analysis, is the thermal power assumed to be 102 percent of the licensed thermal power at the initiation of the transient?

    6.4 Specify the code that is used to determine the transient LHGR limit relative to the 1 percent plastic strain criterion and fuel centerline melt criterion if this code is not RODEX2.

    6.5 During cycle operations, please describe what surveillances or checks are performed by the licensee to ensure that actual plant operations are within the bounds of the COLR analysis in terms of meeting the 1 percent plastic strain and fuel centerline melt criteria.

6.6 Verify that conformance with the operating limit maximum linear heat generation rate (MLHGR) is performed accounting for channel bow. If not, justify why not.

6.7 Verify that conformance with the operating limit MLHGR is performed accounting for local power range monitor (LPRM) rod power biases. If not, justify why not.

6.8 Verify that conformance with the operating limit MLHGR is performed accounting for part length fuel rods (PLFR) fission gas plena. If not, justify why not.

6.9 Clarify if the relevant rod power histories used in the thermal mechanical analysis come from calculated offline or [xxxxx] power histories. Justify the approach used.

6.10 Verify that the power shapes used in the thermal-mechanical calculations of the operating limit and transient limit are conservative for the plant-specific application. If these shapes are different from the shapes reported in BAW-10247(P)(A), justify why they are different.

6.11 The NRC staff is aware that the transient analysis is performed using offline simulations. For the COLR analysis, verify that the steady state offline cycle analysis used to determine the end of cycle (EOC) axial power shape is conservative relative to the operational flexibility allowed by the flow control window along the licensed thermal power line (LTPL) of the approved operating domain.

6.12 The staff is aware that the transient analysis is performed using offline simulations. Verify that the steady state offline cycle tracking analysis is sufficiently detailed to meet the uncertainty requirements imposed on CASMO-4/MICROBURN-B2, in the SE for EMF-2158(P)(A), which references Tables 2.1 and 2.2 in the licensing topical report. The response should provide operational data to verify the accuracy of the offline analysis against plant-specific axial, radial, and nodal traversing in-core probe (TIP) data.

6.13 Specify the code that is used to determine the transient LHGR during simulated feedwater controller failure (FWCF) events if this code is not XCOBRA-T.

6.14 Describe any differences between the XN-NF-84-105(P)(A) licensing topical report description of XCOBRA-T and the current standard production code version that supports the use of this methodology for modern fuel designs such as ATRIUM-10, the response should address axial geometry changes and modern fuel spacers.

6.15 Describe how gamma smearing, decay heat, and direct energy deposition are treated in XCOBRA-T. Provide justification for any assumptions in the analysis. If historical (non-ATRIUM-10 or non-cycle loading specific) parameters are used to model the event, justify the use of these values.

6.16 If XCOBRA-T or another one-dimensional code is used to perform the FWCF event analysis, justify the appropriateness of the assumption to hold the radial power shape constant. This justification should consider the sensitivity of the local sub-bundle radial pin power distribution to the instantaneous void fraction.

6.17 If XCOBRA-T or another one-dimensional code is used to determine the transient hot rod heat flux, please describe how appropriate fuel rod parameters are determined for the analysis. This discussion should address the following: gap conductance, thermal conductivity, pellet size, and heat capacity. Justify that these parameters are acceptably accurate or conservative.

6.18 If XCOBRA-T or another three-equation thermal hydraulic code is used to perform the FWCF event analysis, please justify the appropriateness of utilizing the [xxxxx]. In particular, pressurization in the FWCF may result in significant changes to the fluid saturation temperature. The code treats these temperature changes as [xxxxx] and may result in changes in the cladding heat flux that are nonphysical relative to the expected behavior based on a more detailed two-fluid representation of the liquid film and vapor fields. Provide detailed transient analyses to demonstrate that the predicted transient peak heat flux is accurately calculated or conservative relative to the limitations in the thermal hydraulic model.

6.19 If XCOBRA-T is the code used to perform the transient LHGR analysis, confirm that the thermal hydraulic conditions simulated during the FWCF event do not exceed the application range of the critical heat flux correlation. If these bounds are exceeded the staff is aware that XCOBRA-T [xxxxx]. If the bounds are exceeded provide justification of the application of the analysis to demonstrate acceptable thermal-mechanical performance.

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Sincerely,
/RA/
Jack Donohew, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458
Enclosure
cc w/encl: See next page

DISTRIBUTION:
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*** Memo dated 10/29/2008**
ADAMS Accession No: ML083190830

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	NRR/DSS/SRXB/BC	NRR/LPL4/BC	NRR/LPL4/PM
NAME	JDonohew	JBurkhardt	GCranston*	MMarkley	JDonohew
DATE	12/19/08	11/21/08	10/29/2008	12/19/08	12/19/08

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